Answer Key

Name: ____________________ **Total Points:** 100

Emp. ID: ____________________ **Points Received:** __________

Date: 10/07/2006 **Grade:** __________

Exam / Quiz Title: OCT2006NRC : October 2006 NRC Exam

Open Reference [ ] Yes [X] No

Reference Material that may be used to support this open reference exam:

1. EOP 5A, 7A, SAG.EOP Charts
2. 2.1.10 Att. 1, 5.3GRID, 2.0.5, 5.7.1 Att. 1
3. T.S 3.9.3, 3.3.3.1, Fig 3.1.7.1/2, 3.5.1, 3.8.3, 5.5.8, T3.11.2, T3.4.1-1, DLCO3.1.4

GUIDELINES

1. Allotted time to complete the exam / quiz is **6/8** Hrs.

2. ALL questions shall be directed to the proctor. Students *shall not* discuss the questions among themselves until all examinees have completed the exam / quiz.

3. Restroom trips are limited and only one examinee at a time may leave the room.

4. Verify that all questions have been answered prior to turning in your exam/quiz.

5. To pass the exam/quiz, you must achieve an overall grade of **80/70%** or greater.

6. After you have completed the exam/quiz, you must sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the exam / quiz. This must be done after you have finished.

I have neither given nor received assistance during the administration of this examination/quiz. (Proctor assistance excluded) All work on this exam is my own.

Examinee Signature: ______________________________

Prepared by: ______________________ First Grader: ______________________

Approved by: ______________________ Second Grader: ______________________ (If required)

Approved by: ______________________ (If required)

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
1	21364	00	06/25/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	5	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320104, Ability to interpret the power to flow map following a partial loss of RR Flow

Related Lessons	
INT0320104	CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010400E0200	Discuss Precautions and Limitations associated with Procedure 2.1.10, Station Power Changes.
INT032010400E030A	Discuss the following as described in Procedure 2.1.10, Station Power Changes: General Guidelines for Station Power Changes.

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295001.AA2.01	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.10 / 43.5 / 45.13) Power/flow map (3.5/3.8)

QUESTION: 1 21364 (point(s))

The plant is operating at 100% power when a lightning strike causes the following:

- ➢ Reactor recirc pump runback
- ➢ DEH shifts to mode 3.

The crew locked the scoop tubes for both RR pumps shortly after the transient began. When conditions stabilize the operator notes the following indications:

- ➢ Core flow is 48 MLBH.
- ➢ Reactor power is 95%.
- ➢ Bypass valve position is 50%.

What action is appropriate?

a. Raise reactor recirc flow.

b. Scram the reactor and enter 2.1.5.

c. Reduce reactor power by inserting control rods.

d. Reduce reactor power by reducing reactor recirc flow.

ANSWER: 1 21364
c. Reduce reactor power by inserting control rods.

Provide the Candidate with the Power to Flow Map (2.1.10).

Explanation:

Procedure 2.1.10 requires that if rod line exceeds 120.8%, take action to reduce rod line to 120.8% or below.
The only action that would reduce the operating rod line is answer c.

Procedure 2.1.10 states that this condition may be corrected by:

Reduce power using recirculation flow per Section 6 and/or control rods per Procedure 10.13.
Notify Reactor Engineering as soon as possible following power reduction.

a. is incorrect because this action would not reduce the rod line.
b. is incorrect because there is no requirement to scram the reactor.
d. is incorrect because this action would not reduce rod line.
Source is Modified 5421

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
2	21365	00	06/05/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0010202, Who is required to be notified for a loss of CW pump gland water.

Related Lessons
COR0010202 OPS Circulating Water

Related Objectives	
COR0010202001030D	Describe the interrelationship between the Circulating Water system and the following: Service Water
COR0010202001060C	Predict the consequences a malfunction of the following would have on the Circulating Water system: AC Power

Related References	
5.3AC480	480 VAC Failures

Related Skills (K/A)	
2.1.14	Knowledge of system status criteria which require the notification of plant personnel (such as Reactivity Management Events). (CFR: 43.5 / 45.12) (2.5/3.3)

QUESTION: 2 21365 (1 point(s))

A plant startup is in progress with power at 25% when a loss of **480V BUS 1E** occurs.

What plant notification is required?

a. Notify Security that numerous outside lights are out.

b. Notify PMIS System Engineer that normal UPS power lost.

c. Notify Maintenance to shut down sensitive Weld Shop equipment.

d. Notify CW system engineer that gland water is lost to the CW pumps.

ANSWER: 2 21365
Answer:

d. Notify CW system engineer that gland water is lost to the CW pump.

Explanation:

With power at 25% at least 1 CW pump is running. The loss of 480V Bus 1E results in a loss of gland water to the operating CW pumps. The Circ Water System Engineer is also notified to provide guidance on circ pump operation without gland water.

Distractors:

a. Is incorrect because the loss of 480V 1E does not cause the loss of outside lighting. This notification is required for the loss of 480V 1A or 480V 1B.
b. is incorrect because normal UPS is not lost.
c. Is incorrect because 480V this notification would only be required if Bus 1B is lost.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
3	21366	00	06/05/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	COR0020702, Determine the extent of the loss of DC power.

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001060B	Describe the interrelationship between the DC Electrical Distribution System and the following: AC Electrical Distribution
COR0020702001060I	Describe the interrelationship between the DC Electrical Distribution System and the following: Core Spray
COR0020702001060J	Describe the interrelationship between the DC Electrical Distribution System and the following: RCIC
COR0020702001060M	Describe the interrelationship between the DC Electrical Distribution System and the following: Reactor Feedwater System
COR0020702001080B	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Components using DC control power (i.e., breakers)
COR0020702001110A	Predict the consequences of the following events on the DC Electrical Distribution System: Loss of AC Electrical Distribution
COR0020702001100E	Briefly describe the following concepts as they apply to DC Electrical Distribution System: Loss of breaker protection due to loss of DC Power

Related References	
5.3DC125	Loss of 125 VDC
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295004.AA2.02	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.10 / 43.5 / 45.13) Extent of partial or complete loss of D.C. power (3.5/3.9)

QUESTION: 3 21366 (1 point(s))

The plant was operating at 50% during power ascension when a DC failure occurs. The following **significant conditions** were noted by the crew:

- ➢ Both RFPTs transfer to MDEM
- ➢ RRMGs runback.
- ➢ Both Startup FCVs fail full open
- ➢ 9-3-1/B-7, CORE SPRAY A LOGIC POWER FAILURE annunciates.
- ➢ 9-4-1/A-3, RCIC LOGIC POWER FAILURE annunciates.
- ➢ The BOP notes that **control power to 4160VAC buses A, C, and E is <u>ON</u>.**

What is the extent of the DC failure?

a. Only AA1 is lost.

b. Only AA2 is lost.

c. Only AA1 and AA2 are lost.

d. 125 VDC Distribution Panel A is lost.

ANSWER: 3 21366

b. Only AA2 is lost.

Explanation:

If AA2 is lost RFPT trip logic relay 30TTS will de-energize (RFPT trip input to RRMG runback circuit) and B narrow range level instrument will fail downscale. If B narrow range is the selected level control, RRMGs runback, and RFPTs transfer to MDEM at current speed. Both Startup FCVs will fail full open due to loss of RPV level signal input. RCIC will not operate. High Level trip on HPCI will not function. Core Spray Pump A, RHR Pump A, and RHR Pump B will not start automatically, but can be started from Control Room. DG-1 will not auto start on High Drywell Pressure or Low Vessel Level. The low pressure permissive opening logic for CS-MO-11A, CS-MO-12A, and RHR-MO-27A will not function ; these valves may be driven closed but may not be opened from the Control Room. RFP A has no trip protection other than mechanical overspeed, can not be tripped from Control Room, and must be tripped locally.

Distractors:

a. is incorrect because loss of AA-1 would not effect RCIC, RHR, or Feedwater.

c. is incorrect because no indication of a loss of AA1 is present and if AA1 were lost control power to 4160 A,C and E would be lost.

d. is incorrect because of the absence of the loss of TIP indication and the continued power to AA1.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
4	21367	00	06/05/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Administrative	Turbine trip effect on feedwater temperature and determine what scrams the reactor.

Related Lessons	
INT0060114	ANTICIPATED OPERATIONAL TRANSIENTS
INT0060119	Anticipated Operational Transients and Special Events

Related Objectives	
INT00601140010200	Given an anticipated operational transient that is regularly analyzed, select an action or actions that will terminate the transient.

Related References	
(B)(5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for

Related Skills (K/A)	
295005.AK2.02	Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: (CFR: 41.7 / 45.8) Feedwater temperature. (2.9/3.0)

QUESTION: 4 21367 (1 point(s))

The plant is operating at 22% reactor power with DEH in mode IV when a turbine trip occurs. How is reactor power affected over the next several minutes? (Assume no operator intervention occurs.)

Reactor power…

a. increases due to increased reactor pressure.

b. increases due to decreased feedwater temperature.

c. decreases due to decreased reactor pressure.

d. decreases due to increased feedwater temperature.

ANSWER: 4 21367

b. increases due to decreased feedwater temperature.

Explanation:

Reactor power is low enough that turbine stop valve closure and turbine control valve fast closure scrams are bypassed. Following the turbine trip feedwater temperature lowers due to the loss of extraction steam. As the colder feedwater enters the reactor power rises and is initially controlled by the bypass valves. As power rises the bypass valves open in order to control pass the increased steam production from the reactor. The increased reactor power and steam flow will cause a slight increase in reactor pressure due to the controller bias on DEH. But the cause of the power increase is due to feedwater temperature reduction.

Distractors:

a. is incorrect because the cause of the increase in reactor power is feedwater temperature reduction.

c. is incorrect because power increases.

d. is incorrect because power increases.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
5	21058	00	08/03/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	5	Multiple Choice	

Topic Area	Description
Systems	COR0022002, RPIS Indication Following Scram

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001050L	Predict the consequences the following would have on the RMCS and/or RPIS: Reactor scram

Related References	
2039	Control Rod Drive Hydraulics
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
295006.AA1.06	Ability to operate and/or monitor the following as they apply to SCRAM: (CFR: 41.7 / 45.6) CRD hydraulic system (3.5/3.6)

QUESTION: 5 21058 (1 point(s))

The plant is operating at power when a reactor scram occurs. The following indications are noted on the full core display:

- All Rod Drift Lights are lit.
- All control Rod full in lights are lit.

The operator selects control Rod 26-27 and notes that the four rod display is blank. The following time line of conditions/events then occur:

- 12:00 - Reactor pressure and scram discharge volume pressure equalize.
- 12:02 - Half scram reset is obtained.
- 12:04 - Full scram reset is obtained.
- 12:06 - Scram Discharge Volume Vent and Drain valves are opened.

When does the four rod display **FIRST** indicate a rod position?

a. 12:00

b. 12:02

c. 12:04

d. 12:06

ANSWER: 5 21058

a. 12:00

When pressure is equalized between the reactor and the scram discharge volume the D/P across the drive pistons is zero and at that point the control rods would no longer be inserted past 00 position reed switch and the display would go to 00.

Distractors:

b. is incorrect because indication is first regained when pressure between the reactor and the discharge volume equalize.

c. is incorrect because indication is first regained when pressure between the reactor and the discharge volume equalize.

d. is incorrect because indication is first regained when pressure between the reactor and the discharge volume equalize.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
6	21372	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	COR0023402, Effect that Remote Shutdown actions have on LLS.

Related Lessons
COR0023402 Alternate Shutdown (LO)

Related Objectives	
COR0023402001020A	Describe the interrelationship between ASD and the following: Nuclear Pressure Relief (NPR) system

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295016.AK2.01	Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.7 / 45.8) Remote shutdown panel: Plant-Specific (4.4*/4.5*)

QUESTION: 6 21372 (1 point(s))

Following a toxic gas event requiring the control room to be abandoned, the following conditions existed:

- The MSIVs are closed
- **BOTH** Low-Low Set valves are cycling

The **ADS ISOLATION** switch in the Alternate Shutdown Room is now placed in **ISOLATE.(**NO other actions have been taken outside the control room)

What is the effect on the Low-Low Set valves?

a. Both Low-Low Set valves **stop** cycling.

b. Both Low-Low Set valves **continue** to cycle.

c. The Low-Low Set valve (71F), which **can** be controlled from the ASD room **continues** to cycle.

d. The Low-Low Set valve (71D) which **cannot** be controlled from the ASD room **continues** to cycle.

ANSWER: 6 21372

d. The Low-Low Set valve (71D) which **cannot** be controlled from the ASD room **continues** to cycle.

Reference: COR0023402 ASD Room

EXPLANATION: Placing the Isolation switch in ISOLATE will prevent operation of the Low-Low set valve operated from the ASD room but does not affect the other Low-Low set valve.

Distractors:

a. is incorrect because Low-Low set valve 71D can continue to cycle.
b. is incorrect because Low-Low set valve 71D can continue to cycle.
c. is incorrect because Low-Low set valve controlled from the ASD room is not be able to cycle on Low-Low set.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
7	21373	00	06/26/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	COR0021902, Loss of REC Effect on Continued Operation of Components/Systems

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001060I	Given a specific REC malfunction, determine the effect on any of the following: RHR pumps
COR0021902001060J	Given a specific REC malfunction, determine the effect on any of the following: RWCU system
COR0021902001060G	Given a specific REC malfunction, determine the effect on any of the following: CRDH system

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295018.AK1.01	Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.8 to 41.10) Effects on component/system operations (3.5/3.6)

QUESTION: 7 21373 (1 point(s))

The plant is at 100% power when a complete loss of REC occurs. The crew scrams the reactor and all control rods insert. Reactor water level is normal and reactor pressure is being controlled with SRVs.

What system/component can continue to operate indefinitely without REC?

a. One CRD pump.

b. RWCU in the blowdown mode.

c. One reactor recirc pump at 20% speed.

d. One RHR pump in suppression pool cooling.

ANSWER: 7 21373

d. One RHR pump in suppression pool cooling.

Explanation:

Following a loss of REC CRD, and RR are secured due to the lack of cooling. The loss of REC prevents the use of RWCU, due to the loss of cooling to the NRHX. A single RHR pump can be operated indefinitely without REC.

Distractors:

a. is incorrect because the CRD pumps are required to be secured when REC is lost.

b. is incorrect because without REC RWCU operation cannot continue.

c. is incorrect because RR pump operation cannot continue without REC.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
8	21374	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y Y

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	COR0011702, Reason for Service Air Isolation

Related Lessons
COR0011702 Plant Air

Related Objectives	
COR0011702001110A	Given plant conditions, determine if any of the following should occur: Service Air isolation

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
295019.AK3.03	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.5 / 45.6) Service air isolations: Plant-Specific. (3.2/3.2)

QUESTION: 8 21374 (1 point(s))

The plant is operating at power when a loss of plant air occurs. Instrument air pressure falls to 75 psig. What **automatic** action occurs and **why** is this automatic action required?

a. PCV-609 closes to isolate the service air header from instrument air to ensure that service air header pressure is preserved during a loss of instrument air.

b. IA-80 MV closes to separate the reactor building critical and non critical headers in order to preserve pressure in the header without the leak.

c. IA-80 MV closes to isolate the non-critical air loads to ensure instrumentation and controls remain operable to safely shutdown and cooldown the reactor.

d. PCV-609 closes to isolate the Service Air header from instrument air to ensure instrumentation and controls remain operable to safely shutdown and cooldown the reactor.

ANSWER: 8 21374
Answer:

d. PCV-609 closes to isolate the Service Air header from instrument air to ensure instrumentation and controls remain operable to safely shutdown and cooldown the reactor.

Explanation:

PCV-609 isolates the Service Air distribution header from the air compressors and receivers on low system pressure. Automatic isolation of the Service Air header, along with the storage capacity of the air receivers, ensures that the Instrument Air header will be available for a safe shutdown and cool down of the reactor.

Distractors:
a. is incorrect because separating the headers in order to preserve the side without a leak would only occur if the leak were in the service air header. When 609 closes the service air header is isolated from the compressors and its pressure is not preserved no matter the circumstance.
b. is incorrect even though the closure of MV-80 does isolate the non critical load it does not do it automatically and the purpose of closing the valve is the preservation of load to safely shutdown and cooldown.
c. Is incorrect because this valve is not automatically closed as specified in the stem..

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
9	16796	02	06/27/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	5	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT032-01-26, CNS Abnormal Procedures (RO) Cooling Water

Related Lessons	
COR0022202	REACTOR RECIRCULATION
INT0320126	CNS Abnormal Procedures (RO) Cooling Water
INT0231002	Pre-Outage Industry Events

Related Objectives	
COR0022202001060G	Given a specific Reactor Recirculation system or the Recirculation Flow Control system malfunction, determine the effect on any of the following: Reactor Vessel Internals (jet pumps, stratification, bottom head drain temperature, pump starts)
INT0320126Q0Q0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).
INT02310020010500	Identify the actions to be taken in response to a loss of shutdown cooling.

Related References	
2.4SDC	Shutdown Cooling Abnormal
(B)(14)	Principles of heat transfer thermodynamics and fluid mechanics.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295021.AK1.02	Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.8 to 41.10) Thermal stratification (3.3/3.4)

QUESTION: 9 16796 (1 point(s))

With the plant shutdown in Mode 4 and RHR loop "B" operating in Shutdown Cooling, the following conditions exist:

- Reactor pressure is 0 psig
- Recirc suction temperature is 170°F
- Reactor water level is 58" (NR)
- RHR pumps "A" and "C" have both motors disconnected from their pumps

"B" RHR Loop develops a leak and a SDC isolation results. Following the isolation the following parameters were noted:

- Reactor pressure is 0 psig
- Recirc suction temperature is 170°F
- Reactor water level is 1" (NR)

What action is required, and why?

a. RPV water level must be raised to > 48" to aid in natural circulation flow and ensure bulk reactor coolant temperature is known.

b. RPV water level must be raised to > 48" in order to maximize reactor coolant contact with RPV metal for enhanced heat transfer to the Drywell atmosphere.

c. A Reactor Recirculation pump must be started to reduce the possibility of excessive thermal stresses on the CRD stub tubes.

d. A Reactor Recirculation pump must be started in order to reduce the possibility of thermal binding of the RHR-MO-25A/B valves caused by the expected coolant heatup.

ANSWER: 9 16796

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

Explanation:

If the conditions following the SDC isolation persist, the vessel will become thermally stratified. If this occurs bulk coolant temperature may not be known. Which could allow temperature to increase to the point that mode changes without the knowledge of the operators.

<u>Answer source:</u> 2.4SDC p. 10, Attachment 2, step 1.1

Distractors:

b. Water level is raised to enhance natural circulation flow, not heat transfer.

c. A Recirculation pump is started, but not to prevent thermal stresses on the stub tubes. Starting a recirc pump causes thermal stress on the stub tubes.

d. A Recirculation pump is started, but not to prevent thermal binding of the gate valves. These valves are cycled if closed during a cooldown.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
10	21376	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0012102 , Refueling interlock and reason for the interlock.

Related Lessons
COR0012102 Refueling

Related Objectives	
COR0012102001100A	Given conditions associated with refueling activities, determine if the following should occur: Refueling platform (bridge) movement restrictions
COR0012102001100B	Given conditions associated with refueling activities, determine if the following should occur: Refueling mast restrictions

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
295023.AK3.02	Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS: (CFR: 41.5 / 45.6) Interlocks associated with fuel handling equipment (3.4/3.8)

QUESTION: 10 21376 (1 point(s))

Fuel Handling operations are in progress with the Reactor Mode Switch is in REFUEL.

The following sequence of events occurs:

00:00 - Drive the refueling bridge from the fuel pool to over core location 26-27.
00:05 - Lower the Mast from the full up position to the bundle at location 26-27.
00:10 - Grapple the fuel bundle.
00:15 - Raise the mast and fuel bundle.

When is a rod withdrawal block **first received** and what is the reason this rod block is required?

a. 00:05 to prevent inadvertent criticality.

b. 00:15 to prevent inadvertent criticality.

c. 00:05 to prevent unloading a cell with a withdrawn control rod.

d. 00:15 to prevent unloading a cell with a withdrawn control rod.

ANSWER: 10 21376

Answer:

a. 00:05 to prevent inadvertent criticality.

Explanation:

The purpose of refueling interlocks is to restrict control rod movement, and refueling equipment operation, to reinforce operational procedures that prevent making the reactor critical during refueling. A rod block results whenever any of the following groups of conditions are satisfied.

a. If the is in START-UP and the refueling platform is near or over the core.

b. If the Mode switch is in REFUEL and;

1) A second rod is selected for withdrawal when all rods are not full in, or
2) The refueling platform is near or over the core and one or more of the following exists:
- monorail mounted hoist loaded.
- frame mounted hoist loaded.
- fuel grapple loaded.
- fuel grapple not full up

Distractors:

b. is incorrect because a rod block first occurs at 00:05.
c. is incorrect because the reason for the interlock is to prevent inadvertent criticality.
d. is incorrect because the reason is to prevent inadvertent criticality and the block first occurs at 00:05.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
11	3302	01	04/16/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0020302, Effect of High Drywell Pressure on Drywell Ventilation

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001130D	Describe the PCIS design features and/or interlocks that provide for the following: Bypassing of selected isolations
COR0020302001130E	Describe the PCIS design features and/or interlocks that provide for the following: Operator action to defeat/reset isolations
COR0020302001170A	Predict the consequences of the following items on Primary containment: LOCA
COR0020302001210C	Given plant conditions, determine if the following should have occurred: Drywell cooling fan trip.

Related References	
5.8.10	Average Drywell Temperature Calculation
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295024.EK2.18	Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: (CFR: 41.7 / 45.8) Ventilation. (3.3/3.4)

QUESTION: 11 3302 (1 point(s))

A small break LOCA has occurred with the following conditions:

- ➢ Reactor water level is +45" (NR).
- ➢ Reactor pressure is 560 psig.
- ➢ Drywell pressure is 3.1 psig.
- ➢ Drywell temperature is 195°F.
- ➢ Drywell FCU control switches are in RUN.

What action(s) (if any) will operate **ALL** available drywell FCUs?

a. **NO** actions are required, all FCUs are running.

b. Start all the FCUs by placing their control switches in OVERRIDE.

c. Start all the FCUs by placing their control switches to OFF and then to RUN.

d. Start all the FCUs by placing their control switches in OVERRIDE and then RUN.

ANSWER: 11 3302

b. Start all the FCUs by placing their control switches in OVERRIDE.

FOILS: a. FCUs have tripped. c. With the switches in RUN the FCU do not operate with a high drywell signal present. d. If the control switches are placed in RUN the FCUs will trip.

REFERENCE: EOP-3A, PR 5.8.10, Containment Text

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
12	21377	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	Monitor RCIC During RCIC Pressure Control with high reactor pressure.

Related Lessons	
COR0021802	OPS Reactor Core Isolation Cooling
INT0320105	SYSTEM OPERATING PROCEDURES

Related Objectives	
INT03201050000500	Given a specific procedure and situation, discuss any associated cautions or notes stated in the procedure
COR0021802001080D	Describe the RCIC system design features and/or interlocks that provide for the following: Prevention of turbine damage
INT0320105000040B	Given a specific procedure, state the associated precautions concerned with the following items: Temperature, Pressure, Power, Flow, Level

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295025.EA1.05	Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.7 / 45.6) RCIC: Plant-Specific. (3.7/3.7)

QUESTION: 12 21377 (1 point(s))

The plant is operating at 100% power when a group 1 isolation occurred. The crew entered EOP1A **recovered level** and placed **RCIC in pressure control**. The following RCIC and plant parameters were noted by the operator:

- RCIC Steam Inlet pressure (RCIC-PI-94) indicates 1045 psig and is slowly rising.
- RCIC Pump Discharge pressure (RCIC-PI-93) indicates 1050 psig.
- RCIC Flow (RCIC-FIC-91) indicates 275 gpm and is fluctuating.
- TURB SPEED (RCIC-SI-3067) indicates 5650 rpm and is fluctuating.
- Reactor water level is 35"and slowly lowering.

Ask what action is required?

a. Trip the RCIC turbine.

b. Raise RCIC-FIC-91 setpoint.

c. Place RCIC-FIC-91 to manual.

d. Open RCIC-MO-21, PUMP DISCH TO RX VLV

ANSWER: 12 21377

a. Trip the RCIC turbine.

Explanation:

RCIC Turbine speed is above that which requires an automatic turbine trip. RCIC should be manually tripped to accomplish the action that failed to occur automatically.

Distractors:

b. Is incorrect because a turbine trip is required. This too is an appropriate action if a trip is not required.

c. is incorrect because a turbine trip is required. If turbine speed were lower then this would be a correct answer. Fluctuating flow when speed is less than 75% requires manual operation of the controller.

d. Is incorrect because a turbine trip is required. This action would be selected by the candidate that believes the lower reactor level requires immediate injection.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
13	19285	03	10/15/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080618, DW Spray with NPSH exceeded

Related Lessons
INT0080618 OPS EOP AND SAG GRAPHS AND CAUTIONS

Related Objectives	
INT00806180010200	For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

Related References	
5.8	Emergency Operating Procedures (EOPs)
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.
(B)(8)	Components, capacity, and functions of emergency systems.

Related Skills (K/A)
295026.EK1.01 Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE : (CFR: 41.8 to 41.10) Pump NPSH / 5

QUESTION: 13 19285 (1 point(s))

Drywell Sprays, Torus Sprays and Torus Cooling are in service following a LOCA. Several minutes later, the following conditions exist:

- ➢ Drywell pressure is 5 psig.
- ➢ Drywell temperature is 250°F.
- ➢ Torus pressure is 3 psig.
- ➢ Torus average water temperature is 180°F.
- ➢ Primary containment level is 10 feet.
- ➢ RHR loop A system flow is 8000 gpm (**<u>ONLY RHR pump A is operating</u>**).
- ➢ The Control Room Supervisor has directed that operation of RHR and CS pump remain within NPSH and Vortex limits.

How are Drywell sprays affected (if at all) by this direction and why?

Drywell sprays . . .

a. may continue at current values as no limit is being exceeded.

b. must be reduced since RHR Pump NPSH limit is being exceeded.

c. must be reduced since RHR Pump vortex limit is being exceeded.

d. must be secured because the Drywell Spray Initiation Limit has been exceeded.

ANSWER: 13 19285

b. must be reduced since RHR Pump NPSH limit is being exceeded.

<u>Provide EOP Graphs to the candidate.</u>

The NPSH curve is being exceeded requiring flow to be reduced.

Distractors:
a. Since the SM has directed to remain within the limitations the RHR flow must be reduced.
c. The vortex limit is not be exceeded.
d. Exceeding the DWSIL after initiation does not require securing the drywell sprays.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
14	21379	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	5	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080618, Ability to Monitor Drywell Pressure to Determine When Sprays are allowed.

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL
INT0080618	OPS EOP AND SAG GRAPHS AND CAUTIONS

Related Objectives	
INT00806180010200	For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.
INT00806180010300	Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295028.EA1.04	Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.7 / 45.6) Drywell Pressure (3.9/4.0)

QUESTION: 14 21379 (1 point(s))

The plant experienced a loss of drywell cooling and a small unisolable steam leak into containment. The following parameters were noted:

- Drywell Temperature is 275°F and **<u>slowly rising.</u>**
- Drywell pressure is 5 psig and **<u>slowly rising</u>**.
- Torus spray is in operation.

At what point may drywell sprays **<u>first</u>** be used?

a. immediately.

b. When torus pressure reaches 8 psig.

c. When torus pressure reaches 9 psig.

d. When torus pressure exceeds 10 psig.

ANSWER: 14 21379

Answer:

b. When torus pressure reaches 8 psig.

<u>Provide EOP graphs to the candidate.</u>

Explanation:

Currently conditions are in the unsafe region of the DWSIL graph. At a drywell temperature of 270°F the drywell may be sprayed when DW pressure exceeds 7 psig. So of the pressures listed 8 psig is when DW sprays may be initiated.

Distractors:
a. is incorrect because conditions are currently in the unsafe region of the DWSIL graph. The candidate that only executes the DW temperature leg of EOP-3A would choose this answer.
c. Is incorrect even though these conditions are in the safe region of the graph it is not the first as asked in the stem of the question.
d. Is incorrect and would be chosen by the candidate that only executed the DW pressure leg of EOP-3A.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
15	21380	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	1	1	3	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, Reason a Scram is Required on Low SP Water Level.

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Related References	
(B)(8)	Components, capacity, and functions of emergency systems.

Related Skills (K/A)	
295030.EK3.06	Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.5 / 45.6) Reactor SCRAM. (3.6/3.8)

QUESTION: 15 21380 (1 point(s))

The plant is operating at power when a large suppression pool leak occurs. All available sources are making up to the suppression pool. 20 minutes later the following parameters were noted:

- Suppression pool level is 9.8 ft and lowering at 1" every 10 minutes.
- Suppression pool temperature is 88°F.

Why is a reactor scram required?

a. Insufficient volume of water exists to dampen the dynamic loads on the suppression pool during LOCA.

b. Insufficient volume of water exists in the suppression pool to absorb the energy from an ADS blowdown.

c. Level is approaching the opening of the downcomers and steam suppression during a LOCA can no longer be assured.

d. Level is approaching the opening of the T quenchers and steam suppression during an ADS blowdown can no longer assured.

ANSWER: 15 21380

c. Level is approaching the opening of the downcomers and steam suppression during a LOCA can no longer be assured.

Explanation:

The RPV is not permitted to remain at pressure or at power if suppression of steam discharged from the RPV cannot be assured. A primary containment elevation of 9.6 feet is the opening of the downcomers, the point at which steam suppression can longer be assured.

Distractors:

a. is incorrect because even though the bottom of the HCTL graph is at 9.6 ft the basis for this is uncovery of the downcomers.
b. Is incorrect because at this level and temperature there is sufficient heat capacity to absorb an ADS blowdown.
d. Is incorrect because the T-quenchers still are covered.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
16	21381	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070507, Plant condition that indicate non-compliance with Technical Specifications.

Related Lessons
INT0070507 CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT00705070010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.6 LCO.

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.1.33	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3) (3.4/4.0) **EXAM USE ONLY**

QUESTION: 16 21381 (1 point(s))

The plant was operating at 100% with the RCIC Test Surveillance in progress:

- Average Drywell Air Temperature is 145°F.
- Drywell Pressure is 0.70 psig
- Suppression Pool Level is +5.0"
- Suppression Pool Temperature is 99°F.
- Reactor Water Level is 35"
- Reactor Pressure is 1007 psig.

Entry into the Conditions and Action statements of which Technical Specification LCO is required?

a. 3.6.1.4 Drywell Pressure

b. 3.6.1.5 Drywell Air Temperature

c. 3.6.2.1 Suppression Pool Average Temperature

d. 3.6.2.2 Suppression Pool Water Level

ANSWER: 16 21381

d. 3.6.2.2 Suppression Pool Water Level

Explanation:

The indicated suppression pool level is above the LCO level of 12'11". Therefore entry into 3.6.2.2 is required. The indicated level is the equivalent of 13'2".

Distractors:

a. is incorrect because drywell pressure is below the .75 required for entry into 3.6.1.4.
c. is incorrect because average drywell temperature is below that which requires entry into 3.6.1.5..
d. is incorrect because entry into the LCO is not required during testing that adds heat to the SP.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
17	21382	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	5	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320130, Recognize the Plant Conditions that Require Entry Into 5.2FUEL.

Related Lessons
INT0320130 CNS Abnormal Procedures (RO) High Radiation

Related Objectives	
INT0320130E0E0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.3)
LINK ONLY TO EOP/AOP LESSONS/QUESTIONS	

QUESTION: 17 21382 (1 point(s))

A plant accident occurred that resulted in a loss of all reactor feedwater. The crew entered EOP1A and the following events occurred:

- ➢ 1000 Reactor water level goes below TAF
- ➢ 1005 MSL Radiation Monitors rise from near background to their Hi Setpoint
- ➢ 1010 Drywell Radiation Monitors rises above 250 Rem/hr.
- ➢ 1015 Refuel Floor CAM alarms

When is entry into 5.2FUEL **first** required?

a. 1000

b. 1005

c. 1010

d. 1015

ANSWER: 17 21382

Answer:

b. 1005

Explanation:

The rise in MSL Radiation monitors is unexplained for reasons other than fuel failure. This is an entry condition for 5.2Fuel.

Distractors:

a. is incorrect because although reactor water level less than TAF could precipitate into fuel failure in an of itself it is not an entry condition. And minus other indications of fuel failure entry would be inappropriate.
c. is incorrect because although this is an entry condition the stem asked for FIRST entry condition and the entry is FIRST required when the MSL Rad monitors rise.
d. is incorrect because although this is an entry condition the stem asked for FIRST entry condition and the entry is FIRST required when the MSL Rad monitors rise.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
18	14475	01	07/29/2003	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080610, Actions required during ATWS with Group 1 failure

Related Lessons	
INT0080610	OPS EOP FLOWCHART 7A - RPV LEVEL (FAILURE-TO-SCRAM)

Related Objectives	
INT00806100010800	Given plant conditions and EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM), determine required actions.

Related References	
EOP-6A	EOP Flow Chart 6A
EOP-7A	EOP Flow Chart 7A
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295037.EA2.07	Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.10 / 43.5 / 45.13) Containment conditions/isolations. (4.0/4.2*)

QUESTION: 18 14475 (1 point(s))

The plant is operating at rated power when a main turbine bypass valve fails full open. MSL pressure lowered to 700 psig with reactor power at 65%. The Reactor Operator depressed the Manual Scram pushbuttons, but little rod motion occurred. No other operator actions have been taken. Plant conditions are:

- ➢ Reactor pressure is 680 psig and lowering.
- ➢ Reactor power is 26% and steady.
- ➢ Reactor water level is +48" (NR) and steady.
- ➢ Steam Flow is 2.4 MLBH.
- ➢ MSIVs are OPEN.
- ➢ Torus Temperature is 85°F.

What actions are required ***at this time***?

Place the mode switch to shutdown…

a. close the MSIVs and inhibit ADS.

b. close the MSIVs and initiate SLC.

c. leave MSIVs open and trip recirc pumps.

d. leave MSIVs open, inhibit ADS and bypass low RPV water level interlocks.

ANSWER: 18 14475

a. close the MSIVs and inhibit ADS.

Provide to the candidate; EOP 7A with entry conditions, notes and cautions deleted.

EOP 7A requires the operator to ensure that PCIS isolations 1-7 initiated as required. With main steam line pressure below the setpoint with the mode switch still in RUN, a group 1 isolation should be accomplished. Placing the mode switch to shutdown, and Inhibiting ADS are all appropriate initial additional actions. Clear indications are also provided to the operators by the continued lowering reactor pressure that without MSIV closure, reactor pressure cannot be stabilized.

b. SLC initiation is not required until RMS is in SD, ARI has been initiated, Recirc pumps have been tripped.
c. MSIVs should be closed and recirc pumps are not tripped until ARI has been initiated.

d. MSIVs should be closed and the low level RPV interlocks should not be bypassed at this time.\

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
19	21511	00	08/08/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	COR0011602, Interpret Radiation levels and determine when the MVP trips

Related Lessons
COR0011602 Off Gas

Related Objectives	
COR0011602001060B	Describe the interrelationships between the Off gas system and the following: Process radiation monitoring
COR0011602001130C	Given plant conditions, determine if the following should occur: Mechanical Vacuum Pump trip.

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
295038.EA2.03	Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.10 / 43.5 / 45.13) ?Radiation levels (3.5*/4.3*)

QUESTION: 19 21511 (1 point(s))

A hot plant startup plant startup is in progress with the mechanical vacuum pumps in service. As reactor power increased to the POAH MSL and Off-gas Radiation levels increased.

The following then occurs:

16:00 MSL radiation levels reach their High setpoint.
16:05 Off-Gas Radiation Monitors reach their High setpoint.
16:10 MSL radiation levels reach their High-High setpoint.
16:15 Off-Gas Radiation Monitors reach their High-High setpoint.

When do the mechanical vacuum pumps trip?

a. 16:00

b. 16:05

c. 16:10

d. 16:15

ANSWER: 19 21511

c. 16:10

Explanation:

The MVPs trip on a high Main Steam line radiation signal of greater than 3 times the normal full power background radiation level. This occurred at 16:10

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
20	21149	00	08/07/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	3	1	7	Multiple Choice	

Topic Area	Description
Systems	COR0010502, Will DG CO2 System Actuate

Related Lessons
COR0010502 FIRE PROTECTION SYSTEM

Related Objectives	
COR0010502001110F	Given plant conditions, determine if the following should occur: Initiation of Total Flooding High Pressure CO2 in associated Diesel Generator Room.

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
600000.AK2.01	Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors / detectors and valves (2.6/2.7)

QUESTION: 20 21149 (1 point(s))

All four DG-2 NORMAL-ISOLATE switches are in ISOLATE following a fire induced shutdown from outside the Control Room. The DG-1 and DG-2 Total Flooding High Pressure CO_2 system controls are aligned as follows:

- DG-1 System ABORT SWITCH is in NORMAL.
- DG-2 System MAIN-RESERVE switch is in RESERVE.

A small fire breaks out in the day tank room. The DG-2 day tank room thermal detector actuates and one DG-2 room smoke detector actuates.

How does the system respond?

a. #1 DG cylinders discharge into the #2 DG room.

b. #2 DG cylinders discharge into the #2 DG room.

c. #2 DG cylinders discharge into the #1 DG room.

d. No CO_2 would discharge into either DG room.

ANSWER: 20 21149

a. #1 DG cylinders discharge into the #2 DG room.

With DG-2 in the RESERVE position the DG-2 system uses the DG-1 bottles as long as the DG-1 switch is in NORMAL.

Distractors:

b. is incorrect #1 cylinders discharge into the #2 DG room.

c. is incorrect because CO_2 discharges and #2 DG room HVAC does not trip.

d. is incorrect because CO_2 discharges.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
21	21385	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022202, Implications of Low Reactor Level on RR Pump NPSH

Related Lessons
COR0022202 REACTOR RECIRCULATION

Related Objectives	
COR0022202001050E	Briefly describe the following concepts as they apply to the Reactor Recirculation system or to the Recirculation Flow Control system during normal and reduced forced flow conditions: Indications of Pump Cavitation (including low reactor water level affects on carry under, jet pump NPSH and Recirc pump NPSH)
COR0022202001070E	Predict the consequences a malfunction of the following would have on the Reactor Recirculation system or the Recirculation Flow Control system: Feedwater Flow/Feedwater Flow Inputs (including core inlet subcooling and recirc pump NPSH)
COR0022202001100A	Describe the Reactor Recirculation system and/or Recirculation Flow Control system design features and/or interlocks that provide for the following: Adequate Recirculation Pump NPSH

Related References	
(B)(14)	Principles of heat transfer thermodynamics and fluid mechanics.

Related Skills (K/A)	
295009.AK1.02	Knowledge of the operational implications of the following concepts as they apply to LOW REACTOR WATER LEVEL: (CFR: 41.8 to 41.10) Recirculation pump net positive suction head: Plant- Specific (3.0/3.1)

QUESTION: 21 21385 (1 point(s))

What event would result in the lowest value of available NPSH to the reactor recirculation pumps? (For all options assume the RR pumps are operating).

a. Loss of all feedwater at 20% power.

b. Stuck open relief valve at rated power.

c. Inadvertent RCIC initiation at 75% power.

d. Reactor recirc pump runback to 45% from high power.

ANSWER: 21 21385

a. Loss of all feedwater at 20% power.

Explanation:

NPSH is as a measure of the difference between the saturation pressure and the total pressure felt at the inlet of the pump. The total pressure is made up of two elements; the height of the column of water above the pump, and the amount of subcooling at the pump inlet. If the total pressure at the suction of the pump drops below the required NPSH for the pump, cavitation will occur. Cavitation causes excessive noise, pump vibration and reduction in pumping capacity. This leads to reduced pump efficiency and possible pump damage.

During **low power operations** the significant factor affecting the Recirc pump NPSH is the height of the column of water above the pump, about 57 ft (feedwater subcooling effects though present, are minimal). This provides adequate NPSH to the pumps as long as water level is maintained in the normal operating band. Lowering water level will reduce the NPSH for both the Recirc pump and the jet pumps. Therefore a loss of all feedwater at low power would provide a situation where both the low feedwater flow and the low level contribute to a low value of NPSH to the RR pumps.

Distractors:

b. is incorrect because a stuck open relief valve would in actuality raise the value of available NPSH. Some the steam going through the relief valve would not go through the turbine reducing the amount of extraction steam and feedwater heating.
c. is incorrect because RCIC initiation would provide relatively cool water to the downcomer and therefore would increase NPSH available.
d. is incorrect as the this event would not reduce NPSH as much as the loss of feedwater at low power.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
22	19739	02	02/26/2003	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, Interpret Drywell Pressure to Determine Required Actions.

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.
INT00806130011000	Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

Related References	
5.8	Emergency Operating Procedures (EOPs)
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295012.AA2.02	Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Drywell pressure (3.9/4.1)

QUESTION: 22 19739 (1 point(s))

Given the following conditions:

- ➢ A small coolant leak has occurred in the Drywell.
- ➢ Drywell temperature is 185°F and rising slowly.
- ➢ Drywell pressure is 4.0 psig and rising slowly.

What action is required to control Drywell conditions?

a. Initiate drywell sprays.

b. Operate all available drywell cooling.

c. Vent primary containment with torus vent line.

d. Vent primary containment with drywell vent line.

ANSWER: 22 19739

b. Operate all available Drywell Cooling.

This action is specified in the drywell temperature leg of EOP-3A when drywell temperature cannot be maintained below 150°F.

Answer source: EOP flowchart 3A step DW/T-3

Distractors:

a. Drywell sprays are not permitted with torus pressure at the current 4.0 psig.
c. Venting the torus is not allowed with the LOCA signal present and pressure well below PCPL-A.
d. Venting the drywell is not allowed with the LOCA signal present and pressure well below PCPL-A.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
23	21386	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0060119, Reason for a Scram During an Inadvertent Reactivity Addition

Related Lessons	
INT0060114	ANTICIPATED OPERATIONAL TRANSIENTS
INT0060119	Anticipated Operational Transients and Special Events

Related Objectives	
INT00601140010300	Given a list of transients, select the transient that would most limiting with respect to MCPR considerations.
INT00601190010300	Given a list of Anticipated Operational Transients, select the most limiting transient with respect to MCPR (Minimum Critical Power Ratio).
INT00601140010400	Given a transient and list of reasons, choose the reason the given transient would have MCPR limitations.
INT00601190010400	Given an Anticipated Operational Transient and a list of reasons, select the correct response why the given transient would have MCPR limitations.

Related References	
(B)(1)	Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Related Skills (K/A)	
295014.AK3.01	Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.5 / 45.6) Reactor SCRAM. (4.1*/4.1)

QUESTION: 23 21386 (1 point(s))

The plant is operating at 100% power when a generator load reject occurs with a failure of the turbine bypass valves to open.

What reactor scram signal terminates the transient and why is a scram required?

a. High neutron flux signal scrams the reactor to prevent violating LHGR limit.

b. Emergency trip header pressure signal scrams the reactor to prevent violating the MCPR safety limit.

c. High neutron flux signal scrams the reactor to prevent exceeding the RPV pressure safety limit.

d. Emergency trip header pressure signal scrams the reactor to prevent violation of the RPV pressure safety limit.

ANSWER: 23 21386

b. Emergency trip header pressure signal scrams the reactor to prevent violating the MCPR safety limit.

Explanation:

This transient directly threatens MCPR and the transient is terminated by the emergency trip header pressure scram signal. This is a large positive reactivity addition and the reason for the scram is to prevent exceeding MCPR.

Distractors:

a. is incorrect because the expected scram that terminates this transient is the emergency trip header pressure signal. Since this transient causes a large pressure increase it is expected that the candidate that doesn't fully understand this transient would choose this answer because a large pressure transient is expected.
c. is incorrect because the expected scram that terminates this transient is the emergency trip header pressure signal. Since this transient causes a large pressure increase it is expected that the candidate that doesn't fully understand this transient would choose this answer because a large pressure transient is expected. This transient doesn't threaten the pressure safety limit.
d. is incorrect because the transient doesn't threaten the pressure safety limit.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
24	21387	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020401, Locate local controls and indications during a loss of CRD pumps.

Related Lessons
COR0020401 Control Rod Drive Hydraulics

Related Objectives	
COR0020402001050C	Briefly describe the following concepts as they apply to the CRDH system: Pressure indication
COR0020402001110I	Predict the consequences a malfunction of the following would have on the CRDH systems: CRDH pump trip.
COR00204010010300	State the location of the major system components of the Control Rod Drive System. Component locations and the location of local indications/alarms may not be stated in this text. The ability of the individual to trace system flowpaths and state locations is implied. Specific instances may be covered in the lecture, plant tours and/or OJT.

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
2.1.30	Ability to locate and operate components / including local controls. (CFR: 41.7 / 45.7) (3.9/3.4) **NRC EXAM ONLY**

QUESTION: 24   21387 (1 point(s))

During power operation the operating CRD pump trips and the standby CRD pump cannot be started. Isolation of the Charging Water Header is required.

How is the charging water header isolated?

Where is this valve operated?

a. CRD-29 Charging Water Header Root valve is closed.
   Southeast Reactor Building 903 level.

b. CRD-29 Charging Water Header Root valve is closed.
   Southeast Reactor Building 881 level.

c. CRD-MO-20, DRIVE PRESSURE CONT VALVE, is closed.
   Control Room Panel 9-5.

d. CRD-MO-20, DRIVE PRESSURE CONT VALVE, is closed.
   Southeast Reactor Building 903 level.

ANSWER: 24 21387

a. CRD-29 Charging Water Header Root valve is closed.
   Southeast Reactor Building 903 level.

On the loss of both CRD pumps CRD-29 the CHARGING WATER HEADER ROOT VALVE Located in the reactor building 903 South East is closed. Closing this valve isolates the pressure instrument that sends a signal to the Control Room.

Distractors:

b. is incorrect because the valve is located on the 903 level.
c. is incorrect because closing this valve will not preserve the charging water header pressure.
d. is incorrect because closing this valve will not preserve the charging water header pressure.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
25	21388	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080617, Relationship Between Area Temperature Alarms and Secondary Containment

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010200	State the basis for secondary containment parameter maximum normal operating values (MNO).

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295032.EK2.06	Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: (CFR: 41.7 / 45.8) Area temperature monitoring system (3.3/3.4)

QUESTION: 25 21388 (1 point(s))

What is the significance of the setpoint for the Secondary Containment area high temperature alarms?

a. Secondary Containment equipment required for safe shutdown fails.

b. Indication that a steam leak may be occurring in secondary containment.

c. This is the maximum allowed setting for the leak detection instrumentation.

d. At this temperature personnel access is to secondary containment is precluded.

ANSWER: 25 21388

b. Indication that a steam leak may be occurring in secondary containment.

Explanation:

An area temperature above its maximum normal operating (MNO) value (Table 9) is an indication that steam from a primary system may be discharging into secondary containment. The secondary containment temperature MNO values are based on the alarm set points of selected leak detection temperature instrumentation.

Distractors:

a. is incorrect because this would not occur until the Maximum Safe Operating (MSO) temperature is exceeded.
c. is incorrect because the maximum allowed setting for the leak detection instrumentation coincides with the MSO temperature.
d. is incorrect because personnel access is not necessarily precluded at this temperature.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
26	21316	00	07/30/2005	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022802, Monitor SGT System following Auto initiation from a group 6 isolation.

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001080A	Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation

Related References	
COR0022802	Standby Gas Treatment System
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295035.EA1.02	Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.7 / 45.6) SBGT/FRVS. (3.8/3.8)

QUESTION: 26 21316 (1 point(s))

A Group 6 isolation occurs due to high radiation in the reactor building ventilation exhaust plenum. Following the automatic start of SGT the following parameters and control positions were noted:

- Reactor building pressure is -0.40" H_2O.
- SGT train A Filter train pressure drop is 11" H_2O D/P.
- SGT train B Filter train pressure drop is 11" H_2O D/P.
- SGT-DPCV-546A or SGT-DPCV-546B control switches are in AUTO.

What is the expected position/response of SGT differential pressure control valves (SGT-DPCV-546A or SGT-DPCV-546B) and the SGT fan inlet vortex dampers?

SGT differential pressure control valves...

a. are full open and remain full open and the vortex dampers are full open and remain full open.

b. are full open and remain full open and the vortex dampers close to obtain 10" H_2O D/P across their filter trains.

c. modulate to maintain -0.25" H_2O in the Reactor Building and the vortex dampers are full open and remain full open.

d. modulate to maintain -0.25" H_2O vacuum in the Reactor Building and the vortex dampers close to obtain 10" H_2O D/P across their filter trains.

ANSWER: 26 21316

b. are full open and remain full open and the vortex dampers close to obtain 10" H_2O D/P across their filter trains.

A group 6 isolation signal (the reactor building ventilation exhaust plenum radiation) signal causes the differential pressure control valves (SGT-DPCV-546A or SGT-DPCV-546B) to open and remain open until the signal is reset providing their respective control switches are in AUTO. The fan vortex control system limits air stream flow through each filter train so that total pressure drop across the train remains less than 10" of water D/P.

a. is incorrect because the vortex dampers would not be full open.

c. is incorrect because the D/P control valves are open and remain open until the isolation is reset.

d. is incorrect because the D/P control valves are open and remain open until the isolation is reset.

Source: Direct.

<u>Note: Used on OCT2005NRC EXAM</u>

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
27	21390	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080617, Interpret the cause of the high secondary containment water level.

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010200	State the basis for secondary containment parameter maximum normal operating values (MNO).
INT00806170010300	Explain why the reactor must be shutdown and depressurized if a secondary containment parameter exceeds its maximum safe operating value in 2 or more areas and the primary system is discharging into secondary containment.
INT00806170010400	State the basis for the limits of the maximum safe operating values (MSO) as they apply to personnel protection and equipment operability.
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
295036.EA2.03	Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Cause of the high water level. (3.4/3.8)

QUESTION: 27 21390 (1 point(s))

During Operation a large suppression pool leak occurs. Water levels in the NW and SW quads eventually exceed their maximum safe operating levels. (Assume that the crew is able to maintain normal level in the SP).

What action is **required by EOP-5A**?

a. Shutdown per 2.1.4.

b. Shutdown per 2.1.5.

c. Scram and enter EOP-1A only.

d. Scram and enter EOP-1A and Emergency Depressurize the RPV.

ANSWER: 27 21390

b. Shutdown per 2.1.5.

Provide EOP-5A with the entry conditions and cautions removed.

Explanation:

With two areas greater than Maximum Safe but no primary system discharging into secondary containment a shutdown per 2.1.5 is required by SC-15.

Distractors:

a. is incorrect because a shutdown per 2.1.5 is required. This action would have been appropriate prior to two areas exceeding their maximum safe value.

c. is incorrect because entry into EOP-1A from EOP-5A only occurs when a primary system is discharging into containment.

d. is incorrect because entry into EOP-1A from EOP-5A only occurs when a primary system is discharging into containment and an ED is only required if a primary system is discharging into containment.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
28	21391	00	06/27/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022302, Effect that a malfunction of pressure maintenance will have on the RHR/LPCI.

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001080D	Predict the consequences a malfunction of the following will have on the RHR system: Pressure maintenance system

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
203000.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): (CFR: 41.7 / 45.7) Keep fill system (3.3/3.5)

QUESTION: 28 21391 (1 point(s))

The RHR system is in normal LPCI standby lineup with condensate secured to the Reactor Building Auxiliary Condensate System.

What potential effect would the loss of the Reactor Building Auxiliary Condensate Pump have on the RHR system?

a. Lower NPSH to the RHR pumps.

b. Low shutoff head on LPCI initiation.

c. Water hammer on LPCI system initiation.

d. Low system peak flow rate on LPCI initiation.

ANSWER: 28 21391

c. Water hammer on LPCI system initiation.

Explanation:

The Condensate system normally provides pressure maintenance in order to maintain pressure in the pump discharge piping to prevent water hammer on system startup.

Distractors:

a. is incorrect because the loss of pressure maintenance has no effect on NPSH.

b. is incorrect because once the pump suction is flooded the LPCI shutoff head will be identical. The pump suction will remain flooded irrespective of the status of the keep fill system.

d. is incorrect because system flow rate will not be effected. In fact flow may be higher for a short period of time while the voids in the system fill.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29	21392	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022302, Effect that a lowering reactor water level has on SDC

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001090A	Explain the significance of the following as they apply to a loss of Shutdown Cooling: Reactor water (level, pressure, temperature)

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
205000.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.7 / 45.7) Reactor water level (3.6/3.6)

QUESTION: 29 21392 (1 point(s))

The plant is shutdown with B RHR pump in shutdown cooling with reactor water level at +60" when a loss of coolant accident occurs. Reactor water level drops steadily from its current level to less than -150"(WR).

What is the status of RHR-MO-25B (Inboard Injection Valve) and RHR pump B?

a. RHR-MO-25B is open and RHR pump B is ON.

b. RHR-MO-25B is open and RHR pump B is OFF.

c. RHR-MO-25B is closed and RHR pump B is ON.

d. RHR-MO-25B is closed and RHR pump B is OFF.

ANSWER: 29 21392

d. RHR-MO-25B is closed and RHR pump B is OFF.

Explanation:

With B Loop of RHR in Shutdown cooling the SDC isolation is enabled. That is with RHR-MO-17 and 18 open a group 2 isolation results in the closure of 17 and 18 and RHR-MO-25. When 17 and 18 go closed RHR pump B trips on no suction path. If water level continues to drop to the LPCI initiation setpoint then the RHR pump gets a start signal but the pump trips with the breaker anti pump feature due to the continued presence of the no suction path trip signal. RHR-MO-25 remains closed until the 25 reset pushbutton is depressed.

Distractors:

a. is incorrect because RHR-MO-25 is closed and the pump is OFF. The candidate that believes that the LPCI initiation signal realigns MO-25 and successfully starts MO-25 would choose this answer.
b. is incorrect because RHR-MO-25 is closed. The candidate that is knowledgeable about the anti -pump circuit but not the MO-25 logic would choose this answer.
c. is incorrect the RHR pump is OFF. The candidate that is knowledgeable about the MO-25 logic but not the anti pump logic would choose this answer.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
30	21505	00	08/02/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070506, Knowledge of HPCI Technical Specification LCO

Related Lessons	
INT0070506	OPS Tech. Spec. 3.5, Emergency Core Cooling (ECCS) and Reactor Core Isolation Cooling (RCIC) System

Related Objectives	
INT00705060010100	Given a set of plant conditions, recognize non-compliance with a Section 3.5 LCO.

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.2.22	Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2) (3.4/4.1) **EXAM USE ONLY**

QUESTION: 30 21505 (1 point(s))

What plant conditions would require entry into the conditions and actions of Technical Specification LCO 3.5.1 ECCS Operating?

a. HPCI inoperable with reactor pressure at 450 psig.

b. RCIC inoperable with reactor pressure at 450 psig.

c. One Low Low Set (LLS) is found to inoperable in Mode 1.

d. RHR loop A containment spray valve found stuck closed in Mode 1.

ANSWER: 30 21505

a. HPCI inoperable with reactor pressure at 450 psig.

Explanation:

Since reactor pressure is greater than 150 psig LCO 3.5.1 requires HPCI to be operable.

Distractors:

b. is incorrect because RCIC inoperable required entry into a different LCO 3.5.3.

c. is incorrect because this LLS inoperability requires entry into 3.6.1.6.

d. is incorrect because this requires entry into TRM 3.6.1.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
31	21394	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020602, Ability to control reactor water level with Core Spray from the Control Room

Related Lessons
COR0020602 CORE SPRAY

Related Objectives	
COR0020602001050H	Describe the Core Spray system design features and/or interlocks that provide for the following: Automatic system initiation
COR0020602001080H	Given a Core Spray component manipulation, predict and explain the changes in the following: System lineup
COR0020602001090A	Predict the consequences of the following items on the Core Spray System: Valve closures
COR0020602001090B	Predict the consequences of the following items on the Core Spray System: Pump trips
COR0020602001120B	Given plant conditions, determine if any of the following Core Spray Actions should occur: Pump starts.
COR0020602001120D	Given plant conditions, determine if any of the following Core Spray Actions should occur: Valve reposition.

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
209001.A2.02	Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those...: (CFR: 41.5 / 45.6) Valve closures (3.2/3.2)

QUESTION: 31 21394 (1 point(s))

A LOCA occurs that causes **high drywell pressure and low reactor water level**. The Core Spray subsystem automatically initiates and level is restored to +60"(NR).

- Both CS-MO-12A and 12B are closed.
- Both Core Spray pump control switches are taken to stop and then released.

How is the CS system impacted, if at all, should reactor water level fall to -150"(WR)? (**Assume that the High Drywell Pressure initiation signal for Core Spray remains for the entire from the initial LOCA to present.**)

If water level falls to -150"(WR), what action(s), if any, is/are **required** to inject with Core Spray?

a. Both Core Spray systems remain idle.
Start the Cores Spray pumps only.

b. Both Core Spray systems remain idle.
Start the Core Spray pumps and open CS-MO-12A and 12B.

c. Both Core Spray pumps automatically start.
Open CS-MO-12A and 12B.

d. Both Core Spray pumps automatically start.
No actions are required to inject CS-MO-12A and 12B automatically open.

ANSWER: 31 21394

b. Both Core Spray systems remain idle.
Start the Core Spray pumps and open CS-MO-12A and 12B.

Explanation:

Since the Core Spray initiation signal remains present the system does not restart just because reactor water level drops to less than the initiation setpoint. Therefore in order to inject with core spray the operator must restart the CS pumps and reopen CS-MO-12A and 12B per 2.2.9.

Distractors:

a. is incorrect because just restarting the pumps does not restore injection 12A and 12B must be reopened.

c. is incorrect because the CS pumps do no auto start.

d. is incorrect because the CS pumps do no auto start and 12A and 12B have to be reopened.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
32	21506	00	08/02/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022902, Automatic actions that occur following SLC initiation.

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001050F	Describe the SLC design features and/or interlocks that provide for the below: RWCU isolation

Related References	
(B)(9)	Shielding, isolation, and containment design features, including access limitations.

Related Skills (K/A)	
211000.A3.06	Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: (CFR: 41.7 / 45.7) RWCU system isolation: Plant-Specific (4.0*/4.1*)

QUESTION: 32 21506 (1 point(s))

What automatic actions occur when the Standby Liquid Control injection pump "A" control switch is placed in "START"?

SLC pump A starts, Squib valve 14A fires…

a. and squib valve 14B fires.

b. and RWCU-MOV-MO15, Inboard Isolation Valve closes.

c. and RWCU-MOV-MO18, Outboard Isolation Valve closes.

d. RWCU-MO-MO15 and RWCU-MO-18 Inboard and Outboard Isolation Valves Close.

ANSWER: 32 21506

b. and RWCU-MOV-MO15, Inboard Isolation Valve closes.

Explanation:

RWCU-MO-15 automatically closes with the start of SLC pump 1A.

Distractors:

a. is incorrect because the 14B squib valve is automatically fired from the B SLC pump switch.

c. is incorrect because the RWCU-MO-18 valve does not automatically close upon actuation of the SLC pump 1A switch.

d. is incorrect because the RWCU-MO-18 valve does not automatically close upon actuation of the SLC pump 1A switch.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
33	19035	01	07/03/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0022902, 4160 Bus 1F failure, what is the preferred SLC injection method?

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001100C	Predict the consequences a malfunction of the following would have on the SLC system: A.C.Power

Related References
NONE

Related Skills (K/A)	
211000.K2.01	Knowledge of electrical power supplies to the following: (CFR: 41.7) SBLC pumps (2.9*/3.1*)

QUESTION: 33 19035 (1 point(s))

An ATWS condition exists when a loss of power to Bus 1F occurs. The decision to inject SLC has been made.

Based on this power failure, what is the preferred SLC injection method?

a. RCIC

b. RWCU

c. "A" SLC Pump

d. "B" SLC Pump

ANSWER: 33 19035

d. "B" SLC Pump

Explanation: SLC pump B is still available since it is powered from MCC-S which is powered by 4160 Bus 1G.

Distractors:

a. is incorrect. RCIC is not utilized unless the SLC system is unavailable (EOP 5.8.8).

b. is incorrect. RWCU is not utilized unless the SLC system is unavailable (EOP 5.8.8) and power is lost to one of its isolation valves.

c. is incorrect. A SLC pump has lost power and will not start it is powered by MCC K which is powered by 4160 Bus 1F.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
34	570	3	08/06/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022102 Reactor Protection System

Related Lessons
COR0022102 REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001040A	Describe the RPS design features and/or interlocks that provide for the following: System redundancy and reliability
COR0022102001050A	Briefly describe the following concepts as they apply to RPS: Logic arrangements

Related References
NONE

Related Skills (K/A)	
212000.K5.02	Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: (CFR: 41.5 / 45.3) Specific logic arrangements (3.3/3.4)

QUESTION: 34 570 (1 point(s))

The low water level scram is arranged to provide __________ to ensure reliability and minimize inadvertent trips.

a. one-out-of-two-taken-once logic

b. one-out-of-two-taken-twice logic

c. two-out-of-three-taken-once logic

d. two-out-of-four-taken-twice logic

ANSWER: 34 570

b. one-out-of-two-taken-twice logic

The automatic scram channels contain the redundant contacts and switches which are operated by the separate reactor plant parameter sensors (i.e., each channel contains one contact that is operated by a single pressure switch on high reactor vessel pressure). The automatic scram channels, with their respective trip channels, are aligned to provide the "one-out-of-two taken twice" trip logic of the Reactor Protection System

.Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
35	21510	00	08/06/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0021202, Ability to monitor the effect of operating back panel switches.

Related Lessons
COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives
NONE

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
215003.A4.04	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) IRM back panel switches, meters, and indicating lights (3.1/3.3)

QUESTION: 35 21510 (1 point(s))

In order to support troubleshooting on IRM D, maintenance requires that IRM D mode switch be taken out of operate. None of the planned troubleshooting activities will generate an INOP signal.

What sequence of operation would prevent the generation of an IRM INOP signal?

a. Depress and hold the INOP inhibit pushbutton then take the mode switch out of operate.

b. Take the IRM mode switch out of operate then depress and hold the inop inhibit pushbutton.

c. Momentarily depress the INOP inhibit pushbutton and then take the mode switch out of operate.

d. Take the IRM mode switch out of operate then momentarily depress the inop inhibit pushbutton.

ANSWER: 35 21510

a. Depress and hold the INOP inhibit pushbutton then take the mode switch out of operate.

Explanation:

Taking the mode switch out of operate generates an INOP signal unless the INOP inhibit pushbutton is depressed and held for the entire time the mode switch is out of operate.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
36	1994	01	07/09/2006	10/07/2006	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0023002, SRM detector physical relationship with the RPV.

Related Lessons
COR0023002 SOURCE RANGE MONITOR SUBSYSTEM

Related Objectives	
COR0023002001050A	Given an SRM control manipulation, predict changes in the following: Detector position

Related References	
(B)(2)	General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Related Skills (K/A)	
215004.K1.06	Knowledge of the physical connections and/or cause- effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Reactor vessel (2.8/2.8)

QUESTION: 36 1994 (1 point(s))

Normal operation of the Source Range Monitor detector insert and retract mechanism positions the in-core detector over which of the following ranges?

a. Bottom of active fuel to top of active fuel.

b. Centerline of the core to the bottom of active fuel.

c. 18 inches above the core centerline to 24 inches below the fuel region.

d. 24 inches above core centerline to 24 inches outside the reactor pressure vessel.

ANSWER: 36 1994

c. 18 inches above the core centerline to 24 inches below the fuel region.

The detector has a 10 ft. maximum travel between mechanical limits. The upper mechanical stop is 24 in. above the core midplane. The lower mechanical stop is 24 in. below the active fuel. The detector has approximately a 9.5 ft. travel between the electrical limits (limit switch actuation). The upper electrical stop is 18 in. above the core midplane. This puts the "Full In" position at the same axial position as the centerline of the neutron sources. Therefore, the "Full In" position should be at the point of peak flux during shutdown conditions and while pulling toward criticality. The lower electrical stop is set just above the mechanical stop.

Distractors:

a. is incorrect because the lower limit is 24" below the bottom of the core.

b. is incorrect because the detector upper range is 24" above core midplane

d. is incorrect because the detector is electrically limited to 18 inches above midplane.

REFERENCE: Source Range Monitor Text

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
37	21398	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	5	Multiple Choice	

Topic Area	Description
Systems	COR0020102, Ability to predict the effect of status of rod blocks from APRM

Related Lessons
COR0020102 AVERAGE POWER RANGE MONITOR

Related Objectives	
COR0020102001070A	Given a specific APRM malfunction, determine the effect on any of the following: Reactor Protection System (RPS)
COR0020102001070B	Given a specific APRM malfunction, determine the effect on any of the following: Reactor Manual Control System (RMCS)
COR0020102001080A	Describe the APRM design feature(s) and/or interlock(s) that provide for the following: Rod withdrawal blocks
COR0020102001080B	Describe the APRM design feature(s) and/or interlock(s) that provide for the following: Reactor SCRAM signals

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
215005.A1.03	Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: (CFR: 41.5 / 45.5) Control rod block status (3.6/3.6)

QUESTION: 37 21398 (1 point(s))

The **reactor mode** switch is in **RUN**, and reactor power is 15%. IRMs "B" and "H" failed upscale. Before any operator action is taken APRM "B" and "E" both fail downscale.

What is/are the **minimum** actions to clear **all** rod blocks and/or scrams?

a. Bypass APRM "B".

b. Bypass APRM "E".

c. Bypass APRM "B" and IRM "B".

d. Bypass APRM "B" and APRM "E".

ANSWER: 37 21398

d. Bypass APRM "B" and APRM "E".

Explanation: The upscale IRM B and the downscale APRM B generates a RPS trip on the "B" RPS. Both the downscale APRM "B" and "E" cause a rod block. To clear the RPS trip APRM "B" has to be bypassed. Which still leaves us with a rod block from the downscale APRM "E". Therefore bypassing APRM E bypasses the rod block.

Distractors:

a. is incorrect because a half scram also occurs and to clear all blocks and scrams requires that APRM "E" also be bypassed.

b. is incorrect because the half scram is on RPS "B" and to clear all blocks and scrams requires that APRM "E" also be bypassed.

c. is incorrect because APRM E would still be generating a rod block..

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
38	21399	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0021802, Actions required by RCIC Annunciator

Related Lessons
COR0021802 OPS Reactor Core Isolation Cooling

Related Objectives
NONE

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 38 21399 (1 point(s))

The plant is operating at power with HPCI out of service when a loss of feedwater occurred. RCIC automatically initiated when reactor water level lowered to less than -42". Several minutes later the following indications were present:

- ➢ Reactor Water level is -45".
- ➢ RCIC flow is 250 gpm and steady.
- ➢ RCIC controller is in automatic.
- ➢ Annunciator 9-4-1/A-2 RCIC STEAM LINE HIGH D/P is alarming.

What operator action is required?

a. Depress and Hold the RCIC TRIP pushbutton.

b. Place RCIC flow controller to automatic and raise RCIC flow.

c. Place RCIC TEST SWITCH to TEST and raise RCIC flow with the TEST POTENTIOMETER.

d. Close RCIC-MO-15, INBD STM SUPP ISOL VLV and RCIC-MO-16, OUTBD STM SUPP ISOL VLV.

ANSWER: 38 21399

d. Close RCIC-MO-15, INBD STM SUPP ISOL VLV and RCIC-MO-16, OUTBD STM SUPP ISOL VLV.

Explanation:

Annunciator 9-4-1/A-2 RCIC STEAM LINE HIGH D/P is alarming which indicates that RCIC steam line flow is at or above the high flow isolation setpoint. AP 9-4-1/a-2 lists the automatic action of closing RCIC-MO-15 and 16. Since these actions have not occurred and did not they should be taken by the operator.

Distractors:

a. is incorrect because an isolation is required..
b. is incorrect because indications of a steam leak are present and an isolation is required.
c. is is incorrect because indications of a steam leak are present and an isolation is required.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
39	21494	00	07/29/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0021602, Knowledge of the ADS Design Feature that ensures adequate pneumatics to operate the ADS valves.

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001080D	Predict the consequences a malfunction of the following would have on the NPR system: Air/Nitrogen to ADS valves

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
218000.K4.04	Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Insures adequate air supply to ADS valves: Plant-Specific (3.5/3.6)

QUESTION: 39 21494 (1 point(s))

Immediately following a simultaneous **loss of Nitrogen and a loss of instrument air** how many times, if any, can an **ADS valve** be actuated from the pneumatics stored in its accumulator if **drywell pressure is at atmospheric pressure**? Assume that reactor pressure is and remains near normal operating pressure).

a. None

b. 2 times

c. 5 times

d. 14 times

ANSWER: 39 21494

c. 5 times\

Explanation:

In the event of a failure of the Instrument Air/Nitrogen Supply, the six relief valves associated with the ADS have accumulators with sufficient capacity to operate their respective valves two times at 70% design Drywell pressure. **This equates to five operations at atmospheric pressure.** The two relief valves associated with the LLS have larger accumulators with sufficient capacity to operate their respective valves fourteen times.

Distractors:

a. is incorrect because the accumulators are protected by check valves and the loss of pneumatic supply will no result in the immediate depressurization of the accumulators.

b. is incorrect because at atmospheric pressure in the drywell the valves can be cycled 5 times.

c. is incorrect because this is the value for the LLS accumulators.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
40	5608	01	03/21/2003	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0021602, Knowledge of the power supply to ADS logic.

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001020A	State the electrical power supply to the following NPR components: ADS logic
COR0021602001080F	Predict the consequences a malfunction of the following would have on the NPR system: D.C. power

Related References	
2.2.1	Nuclear Pressure Relief System
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
791E253	Automatic Blowdown System

Related Skills (K/A)	
218000.K2.01	Knowledge of electrical power supplies to the following: (CFR: 41.7) ADS logic (3.1*/3.3*)

QUESTION: 40 5608 (1 point(s))

An accident has occurred, resulting in the following conditions:

- Reactor pressure is 720 psig and lowering.
- RPV water level is -120" (WR) and stable.
- Drywell pressure is 6.2 psig and rising.
- 125 VDC panel AA2 is de-energized.

If present conditions continue, how will ADS respond?

ADS valves . . .

a. **fail to open** due to loss of logic power.

b. **fail to open** due to RPV water level conditions not met.

c. are opened by the B logic circuit powered from its **normal** power source.

d. are opened by both logic circuits powered from their **alternate** power sources.

ANSWER: 40 5608

c. are opened by the B logic circuit powered from its **normal** power source.

<u>Answer source:</u> COR002-16-02, p. 21, & p. 22, section 3, p. 41
COR002-16-02 Figures 4 & 5

Distractors:

a. ADS will initiate powered from BB2.
b. ADS will initiate.
d. ADS "A" has no alternate source and is de-energized.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
41	21495	00	07/28/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020302, Ability to reset system isolation.

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001050A	Describe the interrelationship between the Primary Containment system and the following: PCIS
COR0020302001060O	Describe the interrelationship between PCIS and the following: Containment nitrogen inerting
COR0020302001080C	State the electrical power supplies to the following: PCIS logic power

Related References	
(B)(9)	Shielding, isolation, and containment design features, including access limitations.

Related Skills (K/A)	
223002.A4.03	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Reset system isolations (3.6/3.5)

QUESTION: 41 21495 (1 point(s))

Given the following conditions:

- ➢ The plant is operating at 100% power with a Nitrogen Makeup to the Primary Containment in progress
- ➢ A loss of the "B" Reactor Protection System (RPS) Motor Generator Set occurs
- ➢ Torus N2 Supply Isolation Valve MO-1301 **AND** Drywell N2 Supply Isolation Valve MO-1312 closed
- ➢ RPS power has been restored

Which of the following actions restore the Nitrogen makeup flowpath?

a. Reset the group isolation signal with MO-1301 and MO-1312 to switches in OPEN.

b. Positions MO-1301 and MO-1312 to switches to OPEN then reset the group isolation signal.

c. Positions MO-1301 and MO-1312 to switches to CLOSE then to OPEN and then reset the group isolation signal.

d. Reset the group isolation signal then place the MO-1301 and MO-1312 to switches to CLOSE then to OPEN.

ANSWER: 41 21495

d. Reset the group isolation signal then place the MO-1301 and MO-1312 to switches to CLOSE then to OPEN.

EXPLANATION OF ANSWER:
After the isolation is reset, the control switches are positioned to close to reset the valve circuit and then to open to reposition the valve.

REFERENCE: COR0020302

Distractors:
a. is incorrect because the logic to open the valve is not reset until the switches are placed to close after the isolation is reset.
b. is incorrect because the logic to open the valve is not reset until the switches are placed to close after the isolation is reset.
c. is incorrect the valves will not open unless the switch is first positioned to close and then to open after the isolation is reset.

Bank

Cognitive Level 1

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
42	18247	01	09/09/2003	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0021902, REC valve response with loss of NSST, SSST and no pumps in STANDBY

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001110C	Given plant conditions, determine if any of the following should occur: Any REC valve automatic reposition

Related References	
(B)(9)	Shielding, isolation, and containment design features, including access limitations.

Related Skills (K/A)	
223002.K1.19	Knowledge of the physical connections and/or cause- effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Component cooling water systems (2.7/2.9)

QUESTION: 42 18247 (1 point(s))

The plant is operating normally at 100% power with the following conditions:

- REC pumps 1A, 1B, and 1D are running.
- To support I&C repair efforts, NO REC pumps are in standby.
- REC heat exchanger A is in service.

A steam leak occurs in the drywell causing drywell pressure to rises and stabilize at +3.5 psig. The reactor scrams, and immediately following the trip of the main turbine, **a startup station service transformer lockout occurs**.

Which REC valve is expected to be OPEN 2 minutes after the turbine trip?

a. REC-MO-713, Heat Exchanger 1B Outlet.

b. REC-MO-712, Heat Exchanger 1A Outlet.

c. REC-MO-714, South Critical Loop Supply.

d. REC-MO-1329, Augmented Radwaste Supply.

ANSWER: 42 18247

c. REC-MO-714, South Critical Loop Supply.

Following the loss of all offsite power, 4160V bus 1F and1G are de-energized, resulting in a loss of power to MCC-S and MCC-K and the subsequent trip of all running REC pumps. As system pressure degrades following the trip of all REC pumps, REC-MO-712 and 713 are signaled to close on REC Heat exchanger outlet low pressure (61.5 and 59.5 psig) following a 40 second time delay, and REC-MO-1329 is signaled to close when the REC system supply header experience a low pressure of 60.5 psig following a 40 second time delay. Undervoltage on 4160V bus 1F and 1G will start DG-1 and DG-2, and signal breakers 1FS and 1GS to close to power bus 1F and 1G from the Emergency Station Service Transformer. After power is restored from the Emergency Station Service Transformer, all valves signaled to close on low pressure will do so, and the drywell high pressure condition **(PCIS Group 6 channel B signal) will open REC-MO-714 after a 30 second time delay and the auto opening of the REC heat exchanger B Service Water outlet valve (SW-MO-651) to its minimum flow position.,**

A. is incorrect. Because REC-MO-713 will close on low pressure.
B. is incorrect. Because REC-MO-712 will close on low pressure.
D. is incorrect. Because REC-MO-1329 will close on low pressure.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
43	21401	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0021602, Effect that depressing LLS logic pushbuttons has on LLS operation..

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001030J	Describe the interrelationships between the Nuclear Pressure Relief system and the following: RPS (low-low set initiation)
COR0021602001080E	Predict the consequences a malfunction of the following would have on the NPR system: A.C. power

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
239002.A1.04	Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including: (CFR: 41.5 / 45.5) Reactor pressure (3.8/3.8)

QUESTION: 43 21401 (1 point(s))

The plant is operating at power when an inadvertent group 1 isolation occurs. The reactor scrams and RV-71D and RV-71F initially lift to control pressure. RV-71D then cycles lifting at 1015 psig and reseating at 875 psig.

If both LLS reset pushbuttons are depressed now, how does reactor pressure respond? (Assume the group 1 isolation is still present and no manual pressure control actions have occurred.)

Reactor pressure…

a. continues to cycle between 1015 psig and 875 psig.

b. rises to approximately 1080 psig then cycles between 1015 and 875.

c. rises to 1080 psig and then cycles between 1080 psig and 1030 psig.

d. rises to 1100 psig and then cycles between 1100 psig and 1050 psig.

ANSWER: 43 21401

b. rises to approximately 1080 psig then cycles between 1015 and 875.

Explanation:

Following the initial transient LLS armed and commenced controlling reactor pressure. The loss of offsite power would have deenergized both RPS systems. The loss of RPS prevents the resetting of the LLS logic.

Distractors:

a. is incorrect because the logic would reset and would not arm again until pressure reached 1080 psig and the lower set relief opened and rearmed LLS.
c. is incorrect because when RV-71D lifts LLS is rearmed and pressure cycles between 1015 an 875 psig.
d. is incorrect because RV-71D will lift before pressure reaches 1100 psig.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
44	21402	00	06/22/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	2	1	3	Multiple Choice	

Topic Area	Description
Systems	INT0320135, Ability to perform actions immediately required without reference to procedures.

Related Lessons
INT0320135 CNS Abnormal Procedures (RO) - Condensate/Feedwater

Related Objectives	
INT0320135H0H0100	Given plant condition(s), determine from memory any automatic actions listed in the applicable Abnormal/Emergency Procedure(s) which will occur due to the event(s).
INT0320135I0I0100	Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.49	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.0)

QUESTION: 44 21402 (1 point(s))

The plant is operating at rated power when an inadvertent scram occurs. The following conditions are now present:

- RFC-LC-83, MASTER LEVEL CONTROLLER is set at 15 inches.
- Reactor MODE SWITCH is in SHUTDOWN.
- Pressure set is at 926 psig and BPVs are maintaining pressure.
- RFPT control is in AUTOMATIC.
- Reactor water level is at 59 inches (NR) and slowly rising.
- Both RFPs are running.

What action is immediately **required**?

a. Trip one RFPT.

b. Trip both RFPTs.

c. Place RFPT Controllers RFC-MA-84A/B to MAN.

d. Lower RFC-CS-SUMAST, STARTUP MASTER CONTROL LEVEL SETPOINT to 15".

ANSWER: 44 21402

b. Trip both RFPTs.

Reactor water level is greater than the high trip point for the RFPTs and therefore both should be tripped.

Distractors:

a,c and d. are all incorrect as any action other than a turbine trip is inappropriate.

Source Direct Production TM 21268

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
45	21403	00	06/23/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022802, Effect a malfunction of SGT has on Secondary Containment Temperature.

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives
NONE

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
261000.K3.01	Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: (CFR: 41.7 /45.6) Secondary containment and environment differential pressure (3.3/3.6)

QUESTION: 45 21403 (1 point(s))

An accident occurs that involves a steam leak in the reactor building. The following conditions exist:

- Both SGT trains started on an automatic initiation signal.
- SGT fan 1F was placed to OFF then placed in STANDBY.
- Reactor building differential pressure is -0.5" H_2O.
- SGT train 1E flow is 1300 cfm.

A failure of the vortex dampers for SGT 1E Fan causes them to slowly close. No operator action occurs.

How is reactor building differential pressure affected by this failure?

Reactor building Differential pressure falls …

a. to and remains at approximately 0" H_2O.

b. to 0" H_2O then increases and stabilizes with a positive D/P.

c. until reactor building differential pressure is 0.0" H_2O then recovers to approximately -0.5" H_2O.

d. until train flow drops to less than 800 cfm then reactor building d/p recovers to approximately -0.5" H_2O.

ANSWER: 45 21403

d. until train flow drops to less than 800 cfm then reactor building d/p recovers to approximately -0.5" H_2O.

Explanation:

At 800 cfm the SGT in Standby starts and recovers pressure to approximately the value before the failure of the vortex damper.

Distractors:

a. is incorrect because the standby train starts at 800 cfm and recovers pressure. The candidate that believes a trip of the running fan causes the start of the standby train would choose this answer.

b. is incorrect because the standby train starts at 800 cfm and recovers pressure. The candidate that believes reactor building pressure would naturally be positive and that believes that the standby train would fail to start would choose this answer.
c. is incorrect because the standby train starts at 800 cfm and restores reactor building D/P.

Source: Direct PTM 3817

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
46	16489	01	04/14/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0022802, Knowledge of the cause effect relationship between SGT and the process radiation monitoring system.

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001080A	Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation
COR0022802001130A	Given plant conditions, determine if any of the following should occur: SGT automatic initiation

Related References	
(B)(11)	Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Related Skills (K/A)	
261000.K1.08	Knowledge of the physical connections and/or cause- effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Process radiation monitoring system (2.8/3.1)

QUESTION: 46 16489 (1 point(s))

The plant is at full power when, annunciator 9-4-1/E-4, RX BLDG VENT HI-HI RAD alarms.

Radiation Monitor readings are:

- ➢ RMP-RM-452A: 14 mrem/hr
- ➢ RMP-RM-452B: 12 mrem/hr
- ➢ RMP-RM-452C: 8 mrem/hr
- ➢ RMP-RM-452D: 7 mrem/hr

Which one of the following is the effect on the Secondary Containment and why?
(Note: Use actual setpoints.)

a. NOT affected because only the **DIVISION I** logic has actuated.

b. NOT affected because only the **DIVISION II** logic has actuated.

c. Isolates and both SGT systems initiate. There is a start signal from both Divisions.

d. Isolates but only "A" SGT initiates. There is NO start signal from one Division.

ANSWER: 46 16489

c. Isolates and both SGT systems initiate. There is a start signal from both Divisions.

Justification: If RMP-RM-452A or C <u>AND</u> RMP-RM-452B or D exceed 10 mrem/hr, Reactor Building isolates, and the SGT system starts.

REFERENCE: 2.2.73; 1.3.1.2 (logic)
2.3_9-4-1; Set Points

Distracter a: Both Divisions are actuated. The reactor building will isolate and SGT starts.
Distracter b: Both Divisions are actuated. The reactor building will isolate and SGT starts.
Distracter d: Both Divisions are actuated. The reactor building will isolate and both trains of SGT will start.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
47	21404	00	06/23/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0010102, Principle involved in paralleling AC sources.

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001100A	Briefly describe the following concepts as they apply to AC Electrical Distribution System: Principle involved with paralleling two AC sources

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
262001.K5.01	Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: (CFR: 41.5 / 45.3) Principle involved with paralleling two A.C. sources (3.1/3.4)

QUESTION: 47 21404 (1 point(s))

DG2 is running unloaded. The crew intends to parallel DG2 with 4160 1G.

- DG voltage is **4050 VAC**.
- 4160 1G voltage is **4160 VAC**.

SYNCH SWITCH is placed to EG2 and the synchroscope is rotating slowly **in the COUNTER -CLOCKWISE direction**.

What DG2 speed and voltage adjustments if any are required before closing EG2?

a. Raise speed and lower voltage.

b. Raise speed and raise voltage.

c. No speed adjustment required and lower voltage.

d. No speed adjustment required and raise voltage.

ANSWER: 47 21404

b. Raise speed and raise voltage.

Explanation:
DG speed is less than synchronous speed so speed should be raised. DG voltage is lower than running so voltage should be raised.

Distractors:
a. is incorrect because voltage should be raised.

c. is incorrect because speed should be raised and voltage lowered.

d. is incorrect because speed should be raised.

Source: Direct Production TM 21322

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
48	17798	01	06/21/2002	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0010102, Knowledge of the design features that transfer NBPP from preferred power to alternate power.

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001060C	Describe the interrelationship between the AC Electrical Distribution System and the following: No Break Power Supply

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
262002.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.): (CFR: 41.7 / 45.7) A.C. electrical power (2.7/2.9)

QUESTION: 48 17798 (1 point(s))

Plant operating normally at 100% power when annunciator "C-1/A-1, 250 VDC Bus 1A Blown Fuse" alarmed. The CRT alarm message indicates "(3705) Static Inverter 1A feeder".

Then, while investigating the above alarm, annunciators "C-2/C-1, 4160V BUS 1A BKR 1AN lockout" and "C-1/B-6, 4160V BUS 1F BKR 1FA TRIP" alarm.

Which of the following describes the expected impact on the NBPP?

The NBPP is . . .

a. deenergized.

b. energized by DG-1.

c. energized from the inverter.

d. energized by the Emergency Transformer.

ANSWER: 48 17798

d. energized by the Emergency Transformer.

References: 5.3NBPP

Justification: Static Inverter 1A loses power from 250 VDC Switchgear 1A due to the blow fuse. However, when 4160V BUS 1A is deenergized undervoltage on 4160V bus 1F signals breaker 1FS to close, powering bus 1F from the Emergency Station Service Transformer; thus, the alternate AC source to the NBPP from MCC-R is available from 480V switchgear 1F and MCC-K which are powered from 4160V BUS 1F.

Foils: a. is incorrect. Because the NBPP is energized from the alternate AC source. b. is incorrect. Because DG-1 will start on 4160V bus 1F undervoltage; however, breaker EG1 will NOT close automatically unless breaker 1FS fails to close. c. is incorrect. Because Static Inverter 1A has lost power from 250 VDC Switchgear 1A due to the blow fuse.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
49	12518	00	03/07/2001	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020702, Ability to monitor alarms and determine system status.

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001080D	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Battery chargers

Related References	
3058	DC One Line Diagram
2.2.25.1	125 VDC Electrical System (Div 1)
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
263000.A3.01	Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: (CFR: 41.7 / 45.7) Meters, dials, recorders, alarms, and indicating lights (3.2/3.3)

QUESTION: 49 12518 (1 point(s))

The plant is at 100% power with the 125 VDC electrical distribution system aligned for normal lineup. The following occur:

- Annunciator C-4/C-7, 125 VDC BATT CHARGER 1B TROUBLE alarms
- CRT alarm message indicates:
  (3765) 125V DC BATTERY CHARGER 1B DC VOLTAGE HIGH (in and reset)
  (3762) 125V DC BATTERY CHARGER 1B AC VOLTAGE FAILURE.
  (3764) 125V DC BATTERY CHARGER 1B DC VOLTAGE LOW

What is the position (open or closed) of the 125V charger 1B AC input and DC output breakers?

The AC input breaker is…

a. open and the DC output breaker is closed.

b. open and the DC output breaker is open.

c. closed and the DC output breaker is closed.

d. closed and the DC output breaker is open.

ANSWER: 49 12518

a. open and the DC output breaker is closed.

The AC input breaker has tripped open due to battery charger 1B DC voltage high. DC output over voltage causes the AC input breaker on a 125V CHARGER to trip. The DC output breaker does NOT automatically trip open. Neither breaker automatically trips open due to loss of AC power to the chargers.

Distractors:
b. is incorrect because the DC breaker is closed.
c. is incorrect as the AC breaker opens.
d. is incorrect as the AC breaker opens and the DC remains closed.

REFERENCE: ALARM PROCEDURE 2.3_C-4, 2.2.25.2

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
50	21405	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020702, Knowledge of DC breaker Interlocks

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001090B	Describe the DC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Breaker interlocks, permissives, bypasses and crossties

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
263000.K4.01	Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Manual/ automatic transfers of control: Plant- Specific (3.1/3.4)

QUESTION: 50 21405 (1 point(s))

What interlocks/physical controls exist for the HPCI Starter Rack that prevent the HPCI Starter Rack from being supplied simultaneously from its normal and emergency source?

a. Transfer switch design prevents simultaneous supply from both sources.

b. Electrical interlock prevents the simultaneous closure of both supply breakers.

c. Mechanical interlock prevents the simultaneous closure of both supply breakers.

d. Administrative control of a padlock on the breakers prevents simultaneous closure of both supply breakers.

ANSWER: 50 21405

a. Transfer switch design prevents simultaneous supply from both sources.

Explanation:

The transfer switch prevents the simultaneous supply of the HPCI SR from both sources. The transfer switch can only be aligned to one source at a time.

b. is incorrect because no electrical interlock on the breakers prevents their simultaneous closure. During normal transfer both breakers are simultaneously closed
c. is incorrect because no mechanical interlock on the breakers prevents their simultaneous closure. During normal transfer both breakers are simultaneously closed.
d. is incorrect because the padlock is used to prevent the inadvertent transfer to the alternate source. During transfer the padlock is removed and both breakers are closed.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
51	21145	00	09/02/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	5	Multiple Choice	

Topic Area	Description
Systems	COR0020802, Voltage Regulator Failure Effect on ECCS Pumps.

Related Lessons
COR0020802 DIESEL GENERATORS

Related Objectives	
COR0020802001080A	Given a specific Diesel Generator malfunction, determine the effect on any of the following: Emergency Core Cooling Systems

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
264000.K3.01	Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: (CFR: 41.7 / 45.4) Emergency core cooling systems (4.2*/4.4*)

QUESTION: 51 21145 (1 point(s))

The plant was operating when a loss of offsite power and a LOCA occurred.

- Both diesel generators initially start and tie to their respective busses as designed.
- Immediately after EG1 automatically closed with DG voltage at 4160 Volts, DG1 voltage regulator fails causing the generator voltage to lower continuously at 50 volts/second.

If this trend continues at its current rate, how are the RHR A, RHR B and CS A pumps affected?

RHR A, RHR B and CS A pump breakers...

a. open 41 seconds after EG1 closed.

b. open as soon as voltage drops to 3880 volts.

c. remain closed until DG lockout occurs due to loss of field.

d. pump breakers remain closed irrespective of generator status and bus voltage.

ANSWER: 51 21145

a. open 41 seconds after EG1 closed.

Lockout of load-shedding (blocking the trip function of motor breakers due to undervoltage) on the critical bus occurs if the off-site power source is unavailable and the bus is energized from its Diesel Generator. This will preclude spurious undervoltage trips for 41 seconds if diesel generator output voltage drops during sequential loading or due to energizing a loaded bus. The load shedding feature will be reinstated 41 seconds after EG1(EG2) close (to allow completion of the load-sequencing action), but will only recur on first level (2300 VAC) undervoltage conditions.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
52	21406	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0011701, Ability to monitor control room pressure indication and determine system alignment.

Related Lessons
COR0011701 OPS Plant Air COR0011702 Plant Air

Related Objectives	
COR0011701001060A	Describe the operation of the interlocks associated with the following components in the Plant Air System: Station Air Compressors
COR0011702001060D	Predict the consequences the following would have on the Plant Air System: Leak in system

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
300000.A4.01	Ability to manually operate and / or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Pressure gauges (2.6/2.7)

QUESTION: 52 21406 (1 point(s))

The Plant is operating at power with the Plant Air compressors operating in the CCM local mode with only one compressor initially running. The following indication is observed by the operator:

PI-606 (Instrument Air Header Supply Pressure) is observed to initially lower to 65 psig, then rise to 85 psig and stabilizes.

What happens?

a. Only the lead compressor is loaded and running.

b. The lead and the next compressor are loaded and running.

c. The lead, next and lag compressors are running and loaded.

d. The lead, next and lag compressors are running. Only the lead and the next are loaded.

ANSWER: 52 21406

c. The lead, next and lag compressors are running and loaded.

Explanation:

If operating in CCM local mode the lead compressor is set at 100 - 110 psi. The next is set at 95 -105 psi, and last at 90-100 psi. Since the air leak resulted in decreased system pressure the next and lag compressors started and loaded. Since pressure never rose to greater than the setpoint for the lag compressor all remain on and loaded.

Distractors:

a. is incorrect because pressure fell to less than that required to start and load the other compressors.
b. is incorrect because the lag compressor is also running.
d. is incorrect because all are loaded and running.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
53	21407	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	3	1	5	Multiple Choice	

Topic Area	Description
Systems	COR0021902, Loss of REC surge tank level and system automatic actions.

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001040A	Describe the REC design features and/or interlocks that provide for the following: Repositioning of REC Supply Valves to components
COR0021902001050A	Briefly describe the following concepts as they apply to REC: Leak or lowering system pressure during accident and transient conditions
COR0021902001040D	Describe the REC design features and/or interlocks that provide for the following: Isolation of Non-Critical Cooling loops
COR0021902001110A	Given plant conditions, determine if any of the following should occur: Non-Critical loop isolation
COR0021902001110B	Given plant conditions, determine if any of the following should occur: Standby pumps automatic start
COR0021902001110C	Given plant conditions, determine if any of the following should occur: Any REC valve automatic reposition

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
400000.A2.02	Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6) High/low surge tank level (2.8/3.0)

QUESTION: 53 21407 (1 point(s))

The plant is shutdown with B RHR in shutdown cooling, when the following indications/reports are received:

- M-1/A-1, REC SYSTEM LOW PRESSURE
- M-1/A-3, REC SURGE TANK LOW LEVEL
- Report from the Reactor building Station Operator that a large rupture exists on the REC Supply Header at the Tee to REC-MO-1329, AUGMENTED RADWASTE SUPPLY.

If these conditions persist for the next few minutes, what loads, if any, remain lined up to REC? (Assume no group 6 isolation signal is present and no operator actions are taken.)

How is the REC system aligned (either automatically or manually) for these conditions?

a. No loads are currently lined up.
Supply critical loops with REC.

b. Only South Critical loop is currently lined up.
Supply critical loops with Service Water.

c. No loads are currently lined up.
Supply critical loops with Service Water.

d. Only South Critical loop is currently lined up.
Supply critical loops with REC.

ANSWER: 53 21407

d. Only South Critical loop is currently lined up.
Supply REC critical loops with a single REC pump.

Explanation:

The loss of REC pressure will cause REC-MO-700, REC-MO-702, REC-MO-712, REC-MO-713, and REC-MO-1329 to close this isolates all non-critical REC loads. Since RHR B is in service the South critical loop will remain lined up to REC since there is no auto closure of the critical loop valves.

Distractors:

a. is incorrect because the south critical loop remains aligned to REC.

b. is incorrect because the leak is/will be isolated by the automatic action of REC-MO-700, REC-MO-702, REC-MO-712, REC-MO-713, and REC-MO-1329 to closure and now a single REC pump is started to supply the intact critical loops.

c. is incorrect because the south loop remains aligned to REC and SW would not be used in this condition to supply the critical loops.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
54	21408	00	06/27/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020402, Ability to monitor automatic alarms in the CRD hydraulic system.

Related Lessons
COR0020402 CONTROL ROD DRIVE HYDRAULICS

Related Objectives	
COR0020402001040F	Describe the CRDH system design features and/or interlocks that provide for the following: Isolation of scram discharge volumes during scram conditions

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
201001.A3.10	Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: (CFR: 41.7/45/7) Lights and alarms (3.0/2.9)

QUESTION: 54 21408 (1 point(s))

The plant is operating when the south scram discharge volume drain valve fails closed. Leakage past the seat for the scram outlet valve that discharges to the south volume cause level in the volume to increase.

As the level in the scram discharge volume rises to 60 " what alarms/automatic actions occur as level rises to 60"?

a. Rod Block ONLY.

b. South SDIV NOT DRAINED Alarm ONLY.

c. South SDIV NOT DRAINED Alarm and rod block ONLY.

d. South SDIV NOT DRAINED Alarm, rod block and reactor scram.

ANSWER: 54 21408

c. South SDIV NOT DRAINED Alarm and rod block ONLY.

Explanation:

The SDIV not drained alarm comes in at 11.5" and the rod block occurs at 46". Therefore by the time level reaches 60" the alarm and the rod block are in.

Distractors:

a. is incorrect because the SDIV NOT Drained alarm is also in.

b. is incorrect because a rod block is also present.

d. is incorrect because a scram signal is not present.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
55	2109	00	08/12/1999	10/07/2006	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022002001060B Reactor Manual Control System and Rod Position Indication System

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001060B	Given a RMCS control manipulation, predict and explain the response of the following: Rod movement sequence lights

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
201002.A1.03	Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) Rod movement sequence lights (3.0/2.9)

QUESTION: 55 2109 (1 point(s))

Select the sequence in which the following Reactor Manual Control System lamps are expected to be illuminated as a control rod is withdrawn one notch.

Rod…

a. Out Permit
Out
Settle

b. Out Permit
In
Out
Settle

c. In
Out Permit
Out
Settle

d. Out Permit
In
Settle
Out
Settle

ANSWER: 55 2109

b. Out Permit
In
Out
Settle

REFERENCE: Reactor Manual Control System Text

\If there are no RWM or other rod withdraw blocks, withdrawal motion is selected and sealed in for one timer cycle, allowing Unlatch (drive in light), Drive Out(out light) and Settle (Settle light).

Distractors:

a. is incorrect because the drive in is first in sequence to unlatch the rod.

c. is incorrect because the out permit is expected to be energized from the outset.
d. is incorrect because there is no settle between the in and out.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
56	3987	00	11/04/1999	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0020502001120C Control Rod Drive Mechanisms

Related Lessons
COR0020502 CONTROL ROD DRIVE MECHANISM

Related Objectives	
COR0020502001120C	Determine the interrelationships between the CRDMs and the following: Rod Position Indicating System

Related References	
2.4CRD	CRD Trouble

Related Skills (K/A)	
201003.A4.02	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) CRD mechanism position: Plant-Specific (3.5/3.5)

QUESTION: 56 3987 (1 point(s))

Rod 38-27 is being continuously withdrawn from position 38 to position 48.

Which statement below describes the condition that will indicate rod 38-27 is uncoupled?

a. Rod position indication does **NOT** change when the rod is withdrawn.

b. The Red Full Out light on the Full Core Display is extinguished **AND** indicated position on the Four-Rod display is 48.

c. The Green Over-travel light on the Full Core Display is illuminated **AND** indicated position on the Four-Rod display is 48.

d. Position indication is lost (goes blank) on the Four-Rod display **AND** the PMIS Computer indicates 99.

ANSWER: 56 3987

d. Position indication is lost (goes blank) on the Four-Rod display **AND** the PMIS Computer indicates 99.

EXPLANATION OF ANSWER: d. Correct. Since the reed switch at position 52 (Over-travel Out) only provides input to the Over-travel alarm, indicated position goes blank. a. Reed switches are actuated by magnets in the drive piston which operates properly. b. Since the reed switch at position 52 (Over-travel Out) only provides input to the Over-travel alarm, indicated position goes blank. c. The Green light on the Full Core Display is for Full In. Since the reed switch at position 52 (Over-travel Out) only provides input to the Over-travel alarm, indicated position goes blank.

REFERENCE: STCOR0020502 Control Rod Drive Mechanism Rev 7; PR 2.4.1.1.2 Uncoupled Control Rod Page 1 Section 1 Rev 8

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
57	21410	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022202, Predict the effect of a RR pump signal failure and determine the corrective actions.

Related Lessons
COR0022202 REACTOR RECIRCULATION

Related Objectives
NONE

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
202001.A2.06	Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate...: (CFR: 41.5 / 45.6) Inadvertent recirculation flow decrease (3.6/3.8)

QUESTION: 57 21410 (1 point(s))

The plant is operating at 96% power with the following conditions:

- ➢ **BOTH** Recirculation Pump MG sets are operating at 90% speed
- ➢ A failure in the "B" Recirculation Pump Flow control circuit results in a demanded pump speed signal to the scoop tube positioner of 10%.

1) If no operator actions are taken **<u>what is the impact</u>** on B RRMG set speed?

2) What **<u>immediate operator action</u>** is required to mitigate the impact of this failure?

a. 20%
Lock the B RRMG set scoop tube.

b. 45%
Lock the B RRMG set scoop tube.

c. 20%
Attempt to stabilize flow with RRFC-SIC-16B.

d. 45%
Attempt to stabilize flow with RRFC-SIC-16B.

ANSWER: 57 21410

a. 20%
Lock the B RRMG set scoop tube.

Explanation:

The speed control signal failure would reduce RRMG set speed to minimum of 20% where electrical and mechanical stops would prevent further decrease. 2.4RR immediate operator action requires the scoop tube Lock out.

Distractors:

b. is incorrect because with no operator action speed would decrease to 45%.
c. is incorrect because the only appropriate operator response is a scoop tube lockup..
d. is incorrect because with no operator action speed would decrease to 20% and the only appropriate operator response is a scoop tube lockup.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
58	7759	00	05/29/2000	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022202, Knowledge of the interlocks associated with RR pump Speed.

Related Lessons
COR0022202 REACTOR RECIRCULATION

Related Objectives	
COR0022202001100L	Describe the Reactor Recirculation system and/or Recirculation Flow Control system design features and/or interlocks that provide for the following: Recirculation Pump Runback
COR0022202001130B	Given plant conditions, determine if any of the following should occur: Recirculation pump runback to the dual speed limiter.

Related References	
(B)(6)	Design, components, and functions of reactivity control mechanisms and instrumentation.

Related Skills (K/A)	
202002.K4.02	Knowledge of RECIRCULATION FLOW CONTROL SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Recirculation pump speed control: Plant-Specific (3.0/3.0)

QUESTION: 58 7759 (1 point(s))

The plant was operating at 80% power when 'A' RFP tripped. RPV water level lowered to 25"(NR) prior to recovering.

What is the automatic response of the Reactor Recirculation System?

a. RR pump speed lowers to 20%.

b. RR pump speed lowers to 45%.

c. RR pump speed remains constant.

d. RR pump MG set field breakers open.

ANSWER: 58 7759

b. RR pump speed lowers to 45%.

RR pumps run back to 45% with one feed pump tripped and level < 27.5 inches.

a. is incorrect. This limiter requires < 20% total feedwater flow.
c. is incorrect since the 45% limiter is activated.
d. is incorrect this would occur if level reached -42 inches.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
59	21411	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0012002, Action to prevent overpressure of the RWCU system.

Related Lessons
COR0012002 OPS Reactor Water Cleanup

Related Objectives	
COR0012002001090D	Describe the RWCU design features and/or interlocks that provide for the following: Piping over-pressurization protection
COR0012002001130G	Given a RWCU component manipulation, predict and explain the changes in the following parameters: RWCU system pressure

Related References	
(B)(9)	Shielding, isolation, and containment design features, including access limitations.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.49	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.0)

QUESTION: 59 21411 (1 point(s))

The plant was operating at power when a RWCU isolation (Group 3) occurred.

Which of the following actions would the operator take to prevent overpressurization of the RWCU system?

a. Return Isolation Valve, MO-68 is cracked open.

b. Blowdown Flow Control Valve PCV-55 is closed.

c. Demin Suction Bypass Valve MO-74 is cracked open.

d. Drain Valve to Radwaste System MO-57 and Drain Valve to the Condenser MO-56 are both cracked open.

ANSWER: 59 21411

c. Demin Suction Bypass Valve MO-74 is cracked open.

Following a RWCU isolation Procedure 2.1.22 requires that MO-74 be cracked open to prevent over pressurization by mini-purge. CRD purge of RWCU Pump seals can over pressurize the pump and piping following closure of MO-15 or MO-18. Opening MO-74 provides a path for CRD flow around the demins to the Reactor Vessel.

Distractors:

a. MO-68 should already be open and this valve alone would not provide overpressure protection from mini-purge following isolation because a path around the now out of service demineralizers is required.
b. This valve should already be closed, in addition its closure would do nothing to prevent over pressurization of the RWCU piping. FCV-55 closes to protect downstream piping from high pressure or upstream piping from low pressure.
d. These valves should not be opened simultaneously as this could result in a loss of vacuum.

Source: Production TM 19675

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
60	16471	00	07/25/2001	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0021502, Knowledge of the effect that a loss of NBI instruments have on the MT.

Related Lessons
COR0021502 NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001020F	Describe the interrelationships between NBI and the following: Main Turbine/Feedwater
COR0021502001040D	Briefly describe the following concepts as they apply to NBI: Vessel DP measurement
COR0021502001050A	Predict the consequences of the following on the NBI: Detector equalizing valve leaks

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
216000.K3.16	Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER Instrumentation will have on following: (CFR: 41.7 / 45.4) Main turbine (3.0/3.1)

QUESTION: 60 16471 (1 point(s))

The plant is at 100% power when the equalizing valve for NBI-LT-52A (Narrow Range Reactor Water level instrument) is accidentally fully opened by I&C. The instrument was **NOT** isolated prior to opening the equalizing valve.

Assume NO operator actions are taken.

What effect will this have on plant operation?

a. The RFPs and the Main Turbine will trip.

b. Only a low reactor water level alarm is received.

c. Only a high reactor water level alarm is received.

d. Only a ½ scram is received on RPS trip system "A".

ANSWER: 60 16471

a. The RFPs and the Main Turbine will trip.

Two of three instruments are required to satisfy the logic. When the "A" level instrument is equalized, it drains the reference leg. "A" and "C" narrow range instruments share a common reference leg, so both instruments would be affected and read higher than actual level. The two high level trip signals are actuated causing the RFPs and Main Turbine to trip.

REFERENCE: 4.6.1; Attachment 2 - 1.2.1, 1.2.4.3, and 2.1

Distracter b: A full scram is received.
Distracter c: A high level trip occurs.
Distracter d: A full scram is received.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
61	21412	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y Y

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0010602, Knowledge of the relationship between handling the fuel shipping cask and fuel pool ventilation.

Related Lessons
COR0010602 FUEL POOL COOLING AND DEMINERALIZING SYSTEM

Related Objectives	
COR0010602001050F	Describe the interrelationship between the FPC system and the following: Reactor Building Ventilation
COR0010602001120D	Given a Fuel Pool Cooling component manipulation, predict and explain the changes in the following parameters: Fuel Pool level
COR0012102001020E	Describe the interrelationship between reactor refueling and servicing equipment and the following: Spent fuel cask
COR0012102001030A	Given a Reactor Refueling and Servicing Equipment manipulation, predict and explain the changes in the following parameters: Spent fuel pool level

Related References	
(B)(13)	Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Related Skills (K/A)	
234000.K1.09	Knowledge of the physical connections and/or cause- effect relationships between FUEL HANDLING EQUIPMENT and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Fuel pool ventilation: Plant-Specific (2.8/2.9)

QUESTION: 61 21412 (1 point(s))

Why is the fuel shipping cask required to be lowered into the fuel pool at such a slow rate?

a. Protect reactor building integrity by limiting velocity of the cask.

b. Protect fuel storage racks by limiting the initial velocity in the event the fuel shipping cask is dropped.

c. Prevents flooding the fuel pool ventilation ducts by limiting the rate of displacement of fuel pool water.

d. Prevent loss of visibility and limits refuel floor radiation levels by reducing the disturbance of particulate in the fuel pool.

ANSWER: 61 21412

c. Prevents flooding the fuel pool ventilation ducts by limiting the rate of displacement of fuel pool water.

Explanation:

The empty shipping cask is lowered slowly into the fuel pool, to assure that fuel pool ventilation ducts are not flooded. This ensures that the water can be processed fast enough to keep from flooding the ducts.

At Cooper Nuclear Station a Reactor Operator can be in charge of this evolution on the refueling floor. This is not a refueling evolution and does not involve the movement of Special Nuclear Materials. It does involve interaction with plant systems, ie fuel pool cooling and the ventilation system. Therefore **no** 10CFR55.43(b) topic can be linked to this question. However this valid for an RO and can be related to 10CFR55.41(b)13.

Distractors:

a. is incorrect as this is actually the reason for the restricted path mode of operation. Which limits crane movement to 18 fpm and to a path over structural members, but doesn't restrict the rate the cask is lowered into the pool.
b. is incorrect this is why the cask cannot be lifted over the fuel racks.
d. is incorrect although this occurrence is possible it is unlikely and the rational behind the reduced lowering rate is just to limit the rate of displacement to prevent flooding the ventilation ducts.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
62	1651	00	02/28/2003	10/07/2006	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010802, HEATING, VENTILATION AND AIR CONDITIONING

Related Lessons
COR0010802 OPS HEATING, VENTILATION AND AIR CONDITIONING

Related Objectives	
COR0010802001070E	Describe the interrelationship between the Control Room HVAC and the following: Fire protection
COR0010802001120A	Describe the control Room HVAC design features and interlocks that provide for the following: Control room HVAC reconfigurations
COR0010802001140A	Briefly describe the following concepts as they apply to Control Room HVAC: Airborne contamination (e.g., radiological,toxic gas, smoke) control
COR0010802001160D	Predict the consequences a malfunction of the following would have on the Control Room HVAC system: Fire protection
COR0010802001200D	Predict the consequences of the following items on the Control Room HVAC: Initiation/failure of fire protection system

Related References	
COR0010802	Heating, Ventilation, and Air Conditioning
2.2.84	HVAC Main Control Room and Cable Spreading Room
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
290003.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC: (CFR: 41.7 / 45.7) Fire protection: Plant-Specific (2.6/2.8)

QUESTION: 62 1651 (1 point(s))

What automatic actions occur when Smoke Detector SD-1001 associated with the Main Control Room ventilation trips?

a. Running Supply Fan trips, and all Fire/Smoke dampers close.

b. Exhaust Fans BF-C-1B and EF-C-1B trip, and all Fire/Smoke dampers close.

c. Running Supply, Exhaust, and Booster Fans trip, and all Fire/Smoke dampers close.

d. Emergency Booster Fan BF-C-1A starts, damper HV-270-AV closes, and HV-271-AV opens.

ANSWER: 62 1651

a. Running Supply Fan trips, and all Fire/Smoke dampers close.

When actuated, a smoke detector (SD-1001) located in the cable spreading room exhaust duct will trip off the supply fans and close fire smoke dampers AD-1544, AD-1545, AD-1546, AD-1547, AD-1581, and AD-1582.

b. is incorrect because the exhaust fans do not trip.
d. is incorrect because this is not an emergence booster fan start signal.
c. is incorrect because only the supply fans trip.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
63	21414	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020202, Ability to monitor feedwater system valves in the CR.

Related Lessons
COR0020202 OPS CONDENSATE AND FEED

Related Objectives	
COR0020202001120C	Given plant conditions, determine if: Minimum Flow Valves should have repositioned

Related References	
(B)(4)	Secondary coolant and auxiliary systems that affect the facility.

Related Skills (K/A)	
259001.A4.04	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) System valves (3.1/2.9)

QUESTION: 63 21414 (1 point(s))

RFP 1A is being placed in service with flow at 3000 gpm. The minimum flow valve is closed. A failure in the control system for the feed pump occurs. As a result of the failure flow first drops to 1000 gpm for approximately 1 minute then the rises and stabilizes at 2500 gpm.

How did the RFP-1A minimum flow valve respond to this transient?

a. Opened as flow decreased below 2000 gpm and it remains open.

b. Opened as flow decreased below 2000 gpm and re-closed as flow rose above 2000 gpm.

c. Opened as flow decreased below 1250 gpm and it remains open.

d. Opened as flow decreased below 1250 gpm and re-closed as flow rose above 1250 gpm.

ANSWER: 63 21414

a. Opened as flow decreased below 2000 gpm and it remains open.

Each Reactor Feed pump (RFP) is equipped with an air operated minimum flow valve which will automatically open if pump flow decreases to approximately 2000 gpm. These valves have no automatic closing feature and must be closed, using the minimum flow c/s on Panel A, when pump discharge flow is greater than 2000 gpm.

Source New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
64	21415	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0011602, Implications of adding additional charcoal filters in series.

Related Lessons
COR0011602 Off Gas

Related Objectives	
COR0011602001080E	Describe the Off Gas system design feature(s) and/or interlock(s) that provide for the following: Maximizing charcoal bed efficiency
COR0011602001090D	Explain the following Off Gas system related concepts: Charcoal absorption of fission product gases
COR0011602001010O	State the purpose of the following items related to Off Gas system: Charcoal Absorber Beds

Related References	
(B)(5)	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for

Related Skills (K/A)	
271000.K5.08	Knowledge of the operational implications of the following concepts as they apply to OFFGAS SYSTEM: (CFR: 41.7 / 45.4) Charcoal absorption of fission product gases (2.5/2.6)

QUESTION: 64   21415 (1 point(s))

The plant is operating at 100% power with Augmented Off-gas charcoal beds A, B and C in service and D, E and F isolated to install special test equipment.

Following the maintenance charcoal beds D, E and F are place in service in parallel with Charcoal beds A, B and C.

What makes the largest contribution to the decrease in radiation levels at the ERP following this lineup change?

The increased hold up time…

a. of tritium allows for more decay before reaching the ERP.

b. of radioactive noble gases allows for more decay before reaching the ERP.

c. of N-16 allows for virtually all the N-16 to decay before reaching the ERP.

d. of activated corrosion products allows for more decay before reaching the ERP.

ANSWER: 64 21415

b. of radioactive noble gases allows for more decay before reaching the ERP.

Explanation:

Placing the charcoal beds in service in parallel effectively reduces to half the flow rates through the charcoal filters. This would increase the hold up times for noble gases and reduce the ERP release rate.

Distractors:

a. is incorrect because tritium is not a measurable component of the ERP effluent.
d. is incorrect because activation products are primarily particulate and are already effectively filtered by the beds already in service.
c. is incorrect because essentially all the N-16 has decayed before it reaches the charcoal filters. And therefore adding filters would not effect the release rates.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
65	21416	00	06/28/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	1	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0010502 , Knowledge of the power supplies to Fire Detection.

Related Lessons
COR0010501 FIRE PROTECTION SYSTEM COR0010502 FIRE PROTECTION SYSTEM

Related Objectives	
COR0010502001050J	Describe the interrelationships between the Fire Protection system and the following: A.C. power

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
286000.K2.03	Knowledge of electrical power supplies to the following: (CFR: 41.7) Fire detection system: Plant-Specific (2.5*/2.7*)

QUESTION: 65 21416 (1 point(s))

What provides power to Diesel Generator room thermal and smoke detectors?

a. 120VAC from NBPP

b. 120VAC from CPP-1

c. 125VDC from DG1 and DG2

d 250VDC Starter Rack (Turbine Building)

ANSWER: 65 21416

c. 125VDC from DG1 and DG2

Explanation:

DG Cardox System is powered from DG1 and DG2.

Distractors:

a. is incorrect because NBPP does not supply the Cardox detectors.

b. is incorrect because CPP-1 does not supply the Cardox system or detectors.

d. is incorrect because 250 VDC does not supply Cardox. the Cardox system does not supply

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
66	21417	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Administrative	INT0320105, Explain and apply system limits and precautions.

Related Lessons	
INT0320105	SYSTEM OPERATING PROCEDURES

Related Objectives	
INT0320105000030B	Given a specific system operating procedure, state the administrative limits concerned with the following items: Temperature, Pressure, Power, Flow, Level
INT0320105000040B	Given a specific procedure, state the associated precautions concerned with the following items: Temperature, Pressure, Power, Flow, Level
INT0320105000040C	Given a specific procedure, state the associated precautions concerned with the following items: Valve operations
INT03201050000500	Given a specific procedure and situation, discuss any associated cautions or notes stated in the procedure

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.1.32	Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) (3.4/3.8)
NRC EXAM ONLY	

QUESTION: 66 21417 (1 point(s))

The plant is shutdown with the following conditions:

- ➢ B RHR loop is in shutdown cooling.
- ➢ RHR loop flow is 7100 gpm.
- ➢ Both RR pumps are shutdown.
- ➢ Reactor Coolant temperature is 125°F.
- ➢ The fuel from one entire core quadrant has been removed to perform a shroud inspection.

What action is required and why?

a. Reduce RHR system flow to prevent RHR pump runout.

b. Raise RHR system flow to prevent excessive pump vibration.

c. Raise RHR system flow to ensure proper reactor coolant mixing.

d. Reduce RHR system flow to prevent excessive vibration of in-core instruments.

ANSWER: 66 21417

d. Reduce RHR system flow to prevent excessive vibration of in-core instruments.

Explanation:

During extended periods when fuel or blade guides are removed from around dry tubes, flow should not exceed 7000 gpm recirculating or shut down cooling system drive flow. This measure is required to limit excessive flow induced vibration of in-core instrumentation.

Distractors:

a. is incorrect because flow does need to be reduced but flow is below the 7700 gpm where runout is a concern.
b. is incorrect because flow should be reduced this would be a problem with much lower flows. (NRC IN 89-08)
c. is incorrect because flow should be reduced.

Source: Modified 19901

Question Number	Question ID	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
67	19324	00	12/12/2005	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	Multiple Choice	N

Topic Area	Description
Systems	COR0022202,Reactor Recirculation; Local Scoop Tube Operation

Related Lessons
COR0022201 REACTOR RECIRCULATION COR0022202 REACTOR RECIRCULATION

Related Objectives	
COR00222010010800	Demonstrate the ability to locate, in the plant, all local indications associated with the Recirculation System and state the significance of each. Component locations and the location of local indications/alarms may not be stated in this text. The ability of the individual to trace system flowpaths and state locations is implied. Specific instances may be covered in the lecture, plant tours and/or OJT.
COR0022202001080A	Given a Reactor Recirculation system control manipulation, predict and explain the changes in the following parameters: Flow: Core, Jet Pump

Related References	
PR	2.2.68.1

Related Skills (K/A)		ROI	SROI
2.1.30.	Ability to locate and operate components / including local controls. (CFR: 41.7 / 45.7) (3.9/3.4)	3.9	3.4

QUESTION: 67 19178 (1 point(s))

The plant is operating at 75% power when a failure requires local scoop tube operation of Reactor Recirculation Pump "A".

Which of the following describes the procedural requirements to lock out the scoop tube to obtain local control?

a. Scoop Tube System 1 Breaker is opened locally.

b. Scoop Tube System Test Switch is placed in TEST locally.

c. Scoop Tube Lockout A pushbutton on Panel 9-4 is depressed.

d. Scoop Tube external limit switch LS-6 or LS-7 are momentarily operated.

ANSWER:

d. Scoop Tube external limit switch LS-6 or LS-7 are momentarily operated.

Explanation:

Local control of the scoop tube is obtained by activating one of the external limit switches.

a. is incorrect. This would cause a scoop tube lock due to loss of power to the scoop positioner but is not the method used to establish local control.

b. is incorrect. This interrupts power to the motor run circuitry for testing but is not the method for establishing local control.

c. is incorrect. This is how a scoop tube is locked from the control room, but is not the method used for establishing local controls.

REFERENCES: COR0022202 Section IV, B.3; 2.2.68.1 Step 17

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
68	14431	01	06/02/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Administrative	COR0010301, Ability to use the plant communications equipment.

Related Lessons
COR0010301 Communications

Related Objectives	
COR0010301001020A	Identify the function of each of the following major components in the communications system: Plant Telephone system

Related References	
10CFR55.41(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.1.16	Ability to operate plant phone / paging system / and two-way radio. (CFR: 41.10 / 45.12) (2.9/2.8)

QUESTION: 68 14431 (1 point(s))

There has been a complete failure of the PBX telephone system at CNS.

Which of the following describes the operation of the control room bypass telephone?

a. Incoming and outgoing calls can be made on-site only.

b. Incoming and outgoing calls can be made to or from other bypass telephones or off-site numbers.

c. Outgoing calls can be made to other bypass telephones. Incoming call capability is disabled.

d. In-coming calls can be taken from other bypass telephones or from off-site. Outgoing calls can only be made to other bypass telephones.

ANSWER: 68 14431

b. Incoming and outgoing calls can be made to or from other bypass telephones or off-site numbers.

Explanation:

Bypass phones can call each other on-site and can dial numbers off-site as well as receive calls from off-site. In the event of a total system failure of the PBX, power fail transfer relays will provide service automatically to the exchange network for seven designated stations located in key areas (i.e. Access Control, Control Room, CAS, SAS, Plant Manager's office, Senior Managers' suite, Switchboard). These telephones are marked as a <u>bypass</u> <u>telephone</u>. In order to operate these telephones (during a system failure), merely pick up the handset which directly connects you to the exchange network, omit the <u>9</u> <u>access</u> <u>code</u>, and then dial the desired local or long distance number. Incoming calls will also be received on these telephones.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
69	20692	00	05/01/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	5	Multiple Choice	

Topic Area	Description
Administrative	SKL0080302, Authorization to Manipulate Valves and Hang Tags

Related Lessons	
SKL0080305	Tagout General
SKL0080302	OPS Configuration Management - Ops

Related Objectives	
SKL00803020010100	Given a component or system control device, determine who is authorized to manipulate it using Administrative Procedure 0.31 as a guide.
SKL00803020010300	Given a Tagging Order situation, identify any precautions and limitations associated with it.

Related References	
0.31	Equipment Status Control
0.9	Tagout
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.2.13	Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13) (3.6/3.8)
NRC EXAM ONLY	

QUESTION: 69 20692 (1 point(s))

IAC personnel are replacing Level Transmitter NBI-LT-91B this shift. The following valves on Instrument Rack 25-52 are required to be repositioned and tagged to support the instrument replacement:

- NBI-515, LT-91B VALVE MANIFOLD HIGH SIDE SHUTOFF
- NBI-516, LT-91B VALVE MANIFOLD LOW SIDE SHUTOFF
- NBI-517, LT-91B VALVE MANIFOLD EQUALIZER

What combination of personnel may be used to perform the valve manipulations and valve tagging?

a. **Any** qualified Station Operator may reposition the valves and **any** qualified Station Operator may tag the valves.

b. **An** IAC worker qualified to manipulate valves may reposition the valves and **any** IAC worker may tag the valves.

c. **An** IAC personnel qualified to manipulate valves may reposition the valves and **any** qualified Station Operator may tag the valves.

d. **Any** qualified Station Operator may operate the valves **if** IAC workers identify the valves and specify the order of operation of the valves **any** qualified Station Operator may tag the valves.

ANSWER: 69 20692

c. **An** IAC personnel qualified to manipulate valves may reposition the valves and **any** qualified Station Operator may tag the valves.

Procedure 0.9 Tagout, step 2.10 requires that "Tagouts affecting instrumentation on Instrument Rack 25-5, 25-5-1, 25-6, 25-6-1, 25-51, or 25-52 require IAC assistance in identification, manipulation, and specifying order in which tags shall be hung and released. IAC personnel shall perform all valve manipulations on these instrument racks." All three of these valves are on instrument rack 25-52.

Personnel designated to hang and pick up Tagouts shall be a Qualified Operator or shall be certified to Non-OPS Tagging Order Performer. Operations, IAC, and Radiological Department personnel may be required to manipulate valves in accordance with
Procedure 0.31 during performance of Tagouts. The qualified Station Operator May hang the tags.

Distractors:

a. is incorrect because IAC personnel are required to manipulate the valves.

b. is incorrect because this distractor specifies ANY IAC person may hang the tags. The IAC person would have to be qualified to Non-OPS Tagging Order Performer in order to hang the tags.

c. is incorrect because IAC personnel are required to manipulate the valves.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
70	16466	01	03/19/2003	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Procedures	INT0320104, CNS ADMINISTRATIVE PROCEDURES GENERAL OPERATING PROCEDURES

Related Lessons
SKL0080301 REACTIVITY CONTROL

Related Objectives	
INT032010400A0100	Discuss Precautions and Limitations outlined in General Operating Procedure 2.1.1, Startup Procedure.
SKL00803010000200	Briefly discuss the key points of the principles to be followed while controlling reactivity.
SKL0080301000030E	Discuss precautions and requirements during control rod movements associated with: Reactor period
SKL00803010000700	Given plant conditions determine the required action for an emergency power reduction (attachment 7 of procedure 10.13).

Related References	
2.1.1	Startup Procedure
(B)(1)	Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.2.1	Ability to perform pre-startup procedures for the facility / including operating those controls associated with plant equipment that could affect reactivity. (CFR: 45.1) (3.7/3.6)

QUESTION: 70 16466 (1 point(s))

During a xenon free reactor startup and heatup, reactor period was infinity after withdrawing a control rod. The following conditions are present with **NO** control rod movement for the last two (2) minutes:

- ➢ The reactor is on range 5 of the IRMs (rising)
- ➢ Reactor period is +120 seconds (shortening)
- ➢ Reactor coolant temperature is 180°F (rising)

What action is required?

a. Insert control rods in reverse order to make the reactor subcritical.

b. Insert control rods only as necessary to maintain period longer than 50 seconds.

c. Bypass the RWM and insert emergency power reduction rods (10.13 Att. 7) to position 00.

d. Range IRMs as necessary to keep them on scale until the Point of Adding Heat (POAH) is reached.

ANSWER: 70 16466

a. Insert control rods in reverse order to make the reactor subcritical.

Step 2.20 of procedure 2.1.1 states that conservative action ***is required*** whenever an unexpected situation arises with respect to reactivity, criticality, power level, or any other anomalous behavior of reactor core. This conservative action should include rod insertion to reduce power or a reactor scram without hesitation whenever such unanticipated or anomalous behavior is encountered. In this case indicated power is below the POAH yet temperature is rising and even with this negative reactivity feedback power is rising and period is getting shorter. All of which are significant indications of a significant anomaly.

Answer source: 2.1.1 p. 4, step 2.20

Distractors:
a. While the administrative limit for period is 50 seconds, the reactor is currently exhibiting anomalous behavior.
b. This action is not conservative; this action would allow the anomalous reactor behavior to continue.
d. At this point in the startup operation would be below the 80% rod line and the emergency power reduction control rods are not available.
Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
71	19169	01	03/20/2003	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080617, Why do you restart TB Vent while in 5A?

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
(B)(12)	Radiological safety principles and procedures.

Related Skills (K/A)	
2.3.11	Ability to control radiation releases. (CFR: 45.9 / 45.10) (2.7/3.2)

QUESTION: 71 19169 (1 point(s))

What is the basis for restarting building ventilation in the Turbine Building when executing EOP-5A, RADIOACTIVITY RELEASE CONTROL?

Operation of Turbine Building ventilation…

a. maintains equipment availability **AND** assures that radioactivity releases pass through a monitored release point.

b. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

c. maintains equipment availability **AND** assures a minimum amount of radioactivity plates out on turbine building surfaces.

d. preserves personnel accessibility **AND** assures a minimum amount of radioactivity plates out on turbine building surfaces.

ANSWER: 71 19169

b. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

Explanation: Continued personnel access to the turbine building, radwaste and augmented radwaste may be essential for responding to emergencies. These structures are not air tight and radioactivity release inside them would not only limit personnel access, but would eventually lead to an unmonitored ground level release. Operation of ventilation in these structures preserves accessibility, and assures that radioactivity is discharged through an elevated, monitored release point.

Answer source: INT008-06-17 p. 13, section B.1

Distractors:

a. The purpose of restarting Turbine Building ventilation is not to preserve equipment availability.

c. The purpose of restarting Turbine Building ventilation is not to preserve equipment availability nor to minimize deposition of radioactivity in the building.

d. The purpose of restarting Turbine Building ventilation is not to minimize deposition of radioactivity in the building.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
72	21418	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y Y

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Administrative	INT0320115, Knowledge of radiation exposure limits

Related Lessons	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115D0D010I	Discuss the following as described in Rad Protection Procedure 9.ALARA.1, Personnel Dosimetry and Occupational Radiation Exposure Program: Lifetime TEDE Guideline

Related References	
(B)(12)	Radiological safety principles and procedures.

Related Skills (K/A)	
2.3.4	Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10) (2.5/3.1)

QUESTION: 72 21418 (1 point(s))

If you have exceeded your Lifetime TEDE Guideline how much exposure are you allowed during the year at CNS? What authority, if any, may grant extension to this allowed exposure?

a. 1000 mrem
Radiological Manager and Site Vice President may authorize an extension.

b. 0 mrem
No extensions are allowed.

c. 1000 mrem
No Extensions are allowed.

d. 0 mrem
Radiological Manager and Site Vice President may authorize an extension.

ANSWER: 72 21418

c. 1000 mrem
No Extensions are allowed.

Explanation: The Lifetime TEDE Guideline states that NPPD shall normally limit an individual's lifetime TEDE in rem to the individual's age in years. In addition an individual exceeding the lifetime TEDE Guideline will be limited to a TEDE of 1000 mrem and will not be granted an extension.

Distractors:

a. is incorrect even though 1000 mrem are allowed no extension to this dose is allowed.

b. is incorrect because 1000 mrem TEDE is allowed.

d. is incorrect because 1000 mrem TEDE is allowed and now extensions are allowed.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
73	20498	01	03/24/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080605, Dispatching personnel during an emergency

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE
INT0080501	EOP WALKTHROUGHS

Related Objectives	
INT00806050010900	Identify any EOP support procedures addressed in Flowchart 1A and apply any associated special operating instructions or cautions.
INT00805010010100	In the control room/plant/Simulator, demonstrate ability to use the control room systems, instrumentation, controls, and SPDS terminals for operator action statements from the EOP's or EOP Support Procedures.

Related References	
5.8.4	Alternate Injection Subsystems (Table 4)
(B)(12)	Radiological safety principles and procedures.

Related Skills (K/A)	
2.4.39	Knowledge of the RO's responsibilities in emergency plan implementation. (CFR: 45.11) (3.3/3.1)

QUESTION: 73 20498 (1 point(s))

During an emergency, the following conditions exist:

- ➢ TSC not yet operational.
- ➢ DW Radiation Monitors are reading 4050 Rem/hr.
- ➢ Area Radiation Monitor RM-RA-8 (Reactor Building South CRD) is reading 80 mRem/hr (on scale).
- ➢ Area Radiation Monitor RM-RA-9 (Reactor building 903 NE CRD equipment area) is reading 70 mRem/hr (on scale).
- ➢ A radiation survey shows the CRD Drive Water Filter area is 100 mRem/hour.
- ➢ CRD must be lined up as an alternate injection subsystem to maintain adequate core cooling.

In addition to standard RP practices, what additional requirements, if any, must be met to dispatch an operator to perform this task?

The operator can perform this task...

a. with no additional requirements.

b. only if extremity dosimetry is worn.

c. only after the TSC is declared operational.

d. if a survey instrument that monitors radiation dose rates is taken.

ANSWER: 73 20498

d. if a survey instrument that monitors radiation dose rates is taken.

Explanation:

If Station Area Radiation Monitors in travel path and work location of dispatched personnel are alarming, but on-scale, dispatched personnel shall carry a survey instrument capable of monitoring radiation dose rates in travel path and work areas. Dispatched personnel accompanied by a Radiological Protection Technician or Chemistry/Radiological Protection On-Site Availability Technician also satisfies this criteria.

Distractors:

a. is incorrect because when ARM are in alarm a survey instrument or an RP is required.
b. is incorrect because extremity monitoring is not operator may be dispatched without an RP technician if a survey instrument is used.
c. is incorrect because the TSC need not be operational to dispatch the operator.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
74	21419	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320107, Knowledge of RO tasks performed outside the Control Room during emergencies including implications of performing those tasks.

Related Lessons	
INT0320107	ABNORMAL CONDITION PROCEDURES

Related Objectives	
INT03201070000500	Given a specific procedure, explain or analyze any NOTES and CAUTIONS addressed in the procedure
INT03201070000300	Given a specific procedure title, or adequate information of plant conditions and indications, analyze the immediate actions required
INT03201070000400	Given a specific procedure title, appraise the key concepts from the discussion section

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.34	Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications. (CFR: 43.5 / 45.13) (3.8/3.6)

QUESTION: 74 21419 (1 point(s))

The plant was operating when the following occurred:

➢ Control Room is evacuated due to toxic gas intrusion.
➢ The reactor is not scrammed before leaving the Control Room.

How is the reactor scrammed?

What group isolations result when the reactor is scrammed from outside the Control Room?

(Assume no automatic isolations occurred prior to the scram).

a. The breaker for RPS Alternate power and the RPS MG set AC input breaker are opened in RPS MG set rooms. Only group isolations 2, 3 and 6 result.

b. The breaker for RPS Alternate power and the RPS MG set AC input breaker are opened in RPS MG set rooms. Group isolations 1,2,3,6 and 7 result.

c. The breakers for all APRMs in the cable spreading room are tripped. Only group isolations 2, 3 and 6 result.

d. The breakers for all APRMs in the cable spreading room are tripped. Group isolations 1,2,3,6 and 7 result.

ANSWER: 74 21419

b. The breaker for RPS Alternate power and the RPS MG set AC input breaker are opened in RPS MG set rooms. Group isolations 1,2,3,6 and 7 result.

Explanation:

5.1ASD has the Control Building operator open the alternate power supply breakers and the MG set supply breakers in order to scram the reactor. This completely deenergizes RPS and results in group isolations 1,2,3,6 and 7..

Distractors:

a. is incorrect because a group 1 and 7 also occur. The candidate that just believes that the only isolations that occur are due to shrink may pick this answer.
c. is incorrect because the reactor is scrammed by deengizing RPSPP1A and 1B. Deenergizing the APRMs would cause a scram but not as directed by procedure.
d. is incorrect because the reactor is scrammed by deengizing RPSPP1A and 1B. Deenergizing the APRMs would cause a scram but not as directed by procedure..

Source: Modified 4231

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
75	1744	02	07/19/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0022302, Residual Heat Removal System

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001030P	Describe RHR System design feature(s) and/or interlocks which provide for the following: Spray flow cooling
COR0022302001170C	Given plant conditions, determine actions necessary to place RHR in the following flowpaths: Drywell Spray

Related References	
791E261	RHR Elementary Diagram
2.2.69	Residual Heat Removal System
2.2.69.3	RHR Suppression Pool Cooling And Containment Spray
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(B)(10)	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Related Skills (K/A)	
2.4.48	Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12) (3.5/3.8)

QUESTION: 75 1744 (1 point(s))

Following a LOCA, the following conditions are present:

- ➢ Reactor pressure is 700 psig (lowering slowly)
- ➢ RPV water level is +50" (Corrected FZ) and stable.
- ➢ Torus pressure is 11.0 psig (rising slowly)
- ➢ Drywell pressure is 12.0 psig (rising slowly)

What are the actions that are required by the **Drywell Spray initiation logic**, in order to initiate RHR "A" Drywell Sprays?

a. Place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN **only**.

b. Place Containment Cooling Valve Control Permissive switch in MANUAL, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN **only**.

c. Place Containment Cooling 2/3 Core Valve Control Permissive switch in OVERRIDE, place the Containment Cooling Valve Control Permissive switch in MANUAL, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN **only**.

d. Depress Containment Spray Initiation Signal Reset pushbutton, place Containment Cooling 2/3 Core Valve Control Permissive switch in OVERRIDE, place the Containment Cooling Valve Control Permissive switch in MANUAL, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN.

ANSWER: 75 1744

b. Place Containment Cooling Valve Control Permissive switch in MANUAL, then place Drywell Inbd (MO-31A) and Outbd (MO-26A) Spray Valve control switches in OPEN **only**.

Distractors:

a. The permissive switch must be placed in MANUAL.
c. No need to place 2/3 core height in override.
d. No need to place 2/3 core height in override or reset logic.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
76	21420	00	02/28/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	5	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320124, Interpret Plant indications including core flow to determine the plant is in Natural Circulation.

Related Lessons
INT0320124 CNS Abnormal Procedure (RO) Reactor Recirculation

Related Objectives	
INT032012400F0F00	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
295001.AA2.03	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.10 / 43.5 / 45.13) Actual core flow (3.3/3.3)

QUESTION: 76 21420 (1 point(s))

The plant is operating at 50% power and 40 MLBH Core Flow in single loop operation.
RR Loop 1A is isolated. A transient occurs to the B RR loop and the following indications are noted by the crew:

- Recirc Pump Diff. Pressure, DPI-156B falls to 0 psid.
- Recirc Loop 1B Flow FI-159B falls to 0 gpm.
- Recirc pump speed indicates 77%.
- RRMG set drive motor breaker and field breaker are closed.
- Actual Core flow is determined to be 23 MLBH.
- Reactor power falls to 44%.

What is required?

a. Enter 2.4RR only

b. Enter 2.4RXPWR

c. Enter 5.3AC120 for a loss of CDP-1A

d. Enter 2.4RR and enter procedure 2.1.5

ANSWER: 76 21420

d. Enter 2.4RR and enter procedure 2.1.5

Explanation.

The parameters given indicate a sheared shaft for the B RR pump. This plant is now operating at greater than 1% power with both RR pumps essentially tripped. 2.4RR directs that if both pumps are tripped to scram the reactor and enter 2.1.5.

Distractors:
a. Entry into 2.4RR is required but a scram and entry to 2.1.5 is also required.
b. is incorrect because entry into 2.4 RXPWR is not required because the cause of the power reduction is known.
c. is incorrect because a loss of CDP-1A has not occurred.

Source New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
77	19219	02	09/24/2003	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070509, Degraded Grid, SSST/ESST operability

Related Lessons	
INT0070509	OPS Tech. Spec. 3.8, Electrical Power Systems

Related Objectives	
INT00705090010100	Given a set of plant conditions, recognize non-compliance with a Section 3.8 LCO.

Related References	
5.3GRID	Degraded Grid Voltage
3.8.1	AC Sources - Operating
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

Related Skills (K/A)	
2.1.33	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3) (3.4/4.0) **EXAM USE ONLY**

QUESTION: 77 19219 (1 point(s))

The plant is operating at rated power with the following conditions:

- A degraded grid condition exists.
- Voltage on CNS 161 kV bus is 168 KV.
- Voltage on Cooper Cornfield 69 kV bus is 71 KV.
- Security Analysis is not in service.
- Doniphan notifies CNS Control Room of "MVAR on Cooper Generator" alarm.
- The main generator is at + 350 MVAR

What is the status of off-site source operability?

a. **BOTH** off-site sources are OPERABLE.

b. **BOTH** off-site sources are **IN**OPERABLE.

c. The Emergency Transformer is **IN**OPERABLE; the Startup Transformer is OPERABLE.

d. The Emergency Transformer is OPERABLE; the Startup Transformer is **IN**OPERABLE.

ANSWER: 77 19219

b. **BOTH** off-site sources are **IN**OPERABLE.

The 345 KV system is connected to the 161 KV system at CNS and is connected to the 69 KV at Brock. Plant experience has shown that CNS VAR changes directly and immediately affect 69 and 161 KV voltages. 5.3GRID Attachment 1 .a states "If MVAR meter OUT $ 150, and the Security Analysis is not in service or not solving,:
- Declare SSST inoperable and enter appropriate Condition and Required Action of LCO 3.8.1, AC Sources - Operating.
- Declare ESST inoperable and enter appropriate Condition and Required Action of LCO 3.8.1, AC Sources - Operating"

Distractors:

a, c, d **BOTH** off-site sources become **IN**OPERABLE.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
78	21422	00	02/28/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320127, Interpret temperatures and select appropriate procedure.

Related Lessons
INT0320127 CNS Abnormal Procedures (RO) Turbine/Generator

Related Objectives
NONE

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
295018.AA2.01	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.10 / 43.5 / 45.13) Component temperatures (3.3/3.4)

QUESTION: 78 21422 (1 point(s))

The plant is operating at near rated power when the RO reports rising generator hydrogen temperatures. The RO also reports that bearing and oil temperatures are normal and TEC temperature and pressure are normal.

What is required?

a. Enter 2.4TURB only.

b. Enter 2.4GENH2 only.

c. Enter 2.4TEC and 2.4GENH2.

d. Enter 2.4TEC and 2.4TURB.

ANSWER: 78 21422

b. Enter 2.4GENH2 only.

Explanation:

The entry conditions for 2.4GENH2 are:

Abnormal generator H_2 gas temperatures or pressures.
Abnormal generator stator temperatures or vibrations.
Abnormal H_2 seal oil pressures.
Abnormal exciter amps or volts.

Since generator hydrogen temperatures are rising and approaching their alarm setpoint an entry conditions exists.

Since only one component cooled by TEC is affected there is no entry condition for 2.4TEC. No entry condition exists for 2.4TURB.

Distractors:
a. is incorrect because no entry condition exists for 2.4TURB.
c. is incorrect because only 2.4GENH2 should be entered.
d. is incorrect because neither of these procedures should be entered.
Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
79	4171	5	08/09/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Administrative	COR0099900 QUESTION #502

Related Lessons	
INT0320203	CONDUCT OF OPERATIONS PROCEDURES (SRO)
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)
INT0320201	CNS PROCEDURES (SRO)

Related Objectives	
INT03202030000900	Using procedure 2.0.5, identify conditions requiring one or four hour NRC notification. (2.0.5)

Related References	
2.0.5	Reports to NRC Operations Center

Related Skills (K/A)	
2.4.30	Knowledge of which events related to system operations/status should be reported to outside agencies. (CFR 43.5 / 45.11) (2.2/3.6)

QUESTION: 79 4171 (1 point(s))

The plant is operating at 70% power and instructions are to raise to 100% power. A generator fault results in a turbine trip and reactor scram. Groups 2, 3, and 6 low reactor water level isolations are received before level is stabilized. All rods are full in. Reactor level is 35" controlled by RFPs and reactor pressure is 900 psig controlled by turbine bypass valves.

What is the most limiting NRC notification requirement?

a. 1 hour

b. 4 hour

c. 8 hours

d. License Event Report

ANSWER: 79 4171

b. 4 hour

EXPLANATION:

RPS actuation and ESF actuations are 4 hr. Reportable events per 10CFR50.72(b)(2)(iv)(B).

Distractors:

a. is incorrect because a one hour report is not required.
b. is incorrect because an 8 hour report is not required.
d. is incorrect because this is not the limiting report.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
80	19939	00	03/07/2003	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	1	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT00705070010200, CNS TECH SPEC 3.6, CONTAINMENT SYSTEMS

Related Lessons	
INT0070507	CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT00705070010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.6 specification.

Related References	
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.
3.6.2.1	Suppression pool average temperature

Related Skills (K/A)	
295013.AK1.04	Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.8 to 41.10) Complete condensation. (2.9/3.2)

QUESTION: 80 19939 (1 point(s))

The plant is operating at 100% power with suppression pool temperature at 98°F 26 hours after HPCI was secured from testing.

According to Technical Specification Bases what potential consequence is there to operation in this condition?

a. HPCI will not be available during a LOCA due to elevated lube oil temperature.

b. The capacity of the torus-to-drywell vacuum breakers will be exceeded during a LOCA.

c. The reactor building-to-torus vacuum breakers will operate if drywell sprays are initiated during a LOCA.

d. Peak primary containment pressure and temperatures will exceed maximum allowable values during a design basis accident (DBA).

ANSWER: 80 19939

d. Peak primary containment pressure and temperatures do not exceed maximum allowable values during a design basis accident (DBA).

Answer source: Tech Spec bases p. B 3.6.2.1

Distractors:

a. HPCI is normally aligned to the ECST and lube oil is not affected until 140°F.
b. This would be an issue for initiating drywell spray with evaporative cooling and is positively affected by high torus temp, not negatively affected.
c. This would be an issue for initiating drywell spray with evaporative cooling and is positively affected by high torus temp, not negatively affected.

55.43 section(s): (2)

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
81	21518	00	08/17/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	1	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080610, ATWS What actions are required?

Related Lessons	
INT0080610	OPS EOP FLOWCHART 7A - RPV LEVEL (FAILURE-TO-SCRAM)

Related Objectives	
INT00806100010700	Identify any EOP support procedure addressed in Flowchart 7A and apply any associated special operating instructions or cautions.
INT00806100010800	Given plant conditions and EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM), determine required actions.

Related References
EOP FLOWCHART 7A - RPV LEVEL (FAILURE-TO-SCRAM)

Related Skills (K/A)	
2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.3)
LINK ONLY TO EOP/AOP LESSONS/QUESTIONS	

QUESTION: 81 21518 (1 point(s))

A group 1 isolation and an ATWS occurred. The crew intentionally lowered level when level power conditions were met and established a level band of -25" to +50" (FZ). An emergency depressurization is required when HCTL is approached. The crew commenced injection when minimum steam cooling pressure was reached.

- All available outside shroud injection systems are injecting at full flow.
- Reactor water level is -30" (Corrected FZ) and lowering.
- Reactor pressure is 55 psig and steady.
- Torus pressure is 10 psig and steady.
- 20% of the SLC tank has been injected
- ESP 5.8.3 Alternate Rod Insertion Methods is in progress.

What is required?

a. Enter EOP-7B RPV Flooding.

b. Start CS and Enter ESP 5.8.15 Alternate Injection Subsystems.

c. Enter ESP 5.8.2 Alternate Emergency Depressurization Systems.

d. Continue attempts to raise level per ESP 5.8.13 Outside the Shroud Injection Systems.

ANSWER: 81 21518

b. Start CS and Enter ESP 5.8.15 Alternate Injection Subsystems.

Explanation:

After the ED the crew is at step FS/L-17 on EOP-7A. All outside the shroud systems are at maximum and the reactor is depressurized and level continues to lower. Outside the shroud systems are now required. The crew should inject with CS and enter 5.8.15.

Distractors:

a. is incorrect because level is determinate.
c. is incorrect because the reactor is depressurized (within 50 psi of the torus).
d. Is incorrect because with the decreasing level and outside the shroud systems already injecting at maximum additional flow is required.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
82	21426	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	5	Multiple Choice	

Topic Area	Description
Emergency Plan	GEN0030121, Ability to interpret radiation levels and classify the event.

Related Lessons
GEN0030121 EMERGENCY ACTION LEVELS

Related Objectives	
GEN00301210000100	Given a scenario describing plant conditions requiring declaration of an Emergency, the student will identify the correct Emergency Class using the Emergency Planning Implementing Procedures.

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selestion of appropriate procedures during normal, abnormal, and emergency sitituations.

Related Skills (K/A)	
295038.EA2.03	Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.10 / 43.5 / 45.13) ?Radiation levels (3.5*/4.3*)

QUESTION: 82 21426 (1 point(s))

The plant is operating at 100% power when the following timeline of events occur:

10:00	Large LOCA occurs and on the subsequent scram very few rods insert into the core.
10:05	Attempts to start SLC pumps are unsuccessful.
10:10	Reactor Water Level Drops below TAF and cannot be raised.
10:15	Drywell Radiation Monitors rise above 10^4 Rem/hr.
10:20	Drywell pressure is 26 psig and slowly rising.
10:25	Projected Dose at the Site Boundary is 50 mrem TEDE and 100 mrem CDE (thyroid).

When is the declaration of a General Emergency **FIRST** required?

a. 10:10

b. 10:15

c. 10:20

d. 10:25

ANSWER: 82 21426

c. 10:20

Provide the candidate with Procedure 5.7.1.

Explanation:

For this scenario a general emergency is first declared when 2 fission product barriers are lost and the potential exists for the loss of the third. The first barrier is lost with the LOCA and the second is lost (fuel clad) when the DW rad monitors read greater than 2500 rem/hour. At 1020 DW pressure rises to greater than 25 psig which is the potential loss of the third barrier and therefore constitutes a GE.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
83	21427	00	06/24/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320107, Determine the cause of the reactivity addition.

Related Lessons
INT0320107 ABNORMAL CONDITION PROCEDURES

Related Objectives
INT03201070000100 Given a list of symptoms, identify the abnormal condition

Related References	
10CFR55.43 (6)	Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Related Skills (K/A)	
295014.AA2.03	Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.10 / 43.5 / 45.13) Cause of reactivity addition (4.0/4.3*)

QUESTION: 83 21427 (1 point(s))
A plant startup and power ascension is in progress with the following conditions:

- ➢ Reactor power is 15%
- ➢ Main Turbine speed is approaching 1800 rpm.
- ➢ Core Plate D/P is 1.4 psid.
- ➢ Reactor Pressure is 930 psig.

Shortly after the turbine reached 1800 rpm reactor power quickly rose to and stabilized at 20% with the following conditions:

- ➢ Reactor power is 20%
- ➢ Main turbine speed is 1800 rpm and steady.
- ➢ Core Plate D/P is 1.4 psid.
- ➢ Reactor pressure is 931 psig.
- ➢ No annunciation occurred before during or after the transient

What is the cause of the reactor power increase?

What procedure entry is required?

a. Dropped control rod
2.4CRD, CRD Trouble

b. Dropped control rod
2.4RXPWR, Reactor Power Anomalies

c. Control Rod Drifting Out
2.4RXPWR, Reactor Power Anomalies

d. Control Rod Drifting Out
2.4CRD, CRD Trouble

ANSWER: 83 21427
b. Dropped control rod
2.4RXPWR, Reactor Power Anomalies

The cause of the power rise is a dropped control rod. The drifting control rod is ruled out because there is no annunciation. An unexplained increase in reactor power required enty into 2.4RXPWR.

Distractors:

a. 2.4CRD would not be entered for a dropped control rod.

c. Indications are that a rod drop accident occurred. For a rod drop accident 2.4RXPWR is entered.

d. Indications are that a rod drop accident occurred.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
84	21428	00	07/09/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070508, Knowledge of the bases in TS for CREFS actuation.

Related Lessons	
INT0070508	CNS Tech. Spec. 3.7, Plant Systems

Related Objectives	
INT00705080010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.7 Specification.

Related References	
10CFR55.43 (2)	Facility operating limitations in the technial specifications and their bases.

Related Skills (K/A)	
2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CFR: 43.2) (2.5/3.7)
LINK ONLY TO TECH SPEC LESSONS & QUESTIONS	

QUESTION: 84 21428 (1 point(s))

Technical Specifications require that the CREF system be operable during handling of irradiated fuel.

What is the bases for this requirement?

The bases for this requirement is that in the event if a fuel handling accident that releases radioactivity to Secondary containment the dose to control room personnel is limited to …

a. 5 Rem whole body for a 200 man-day continuous occupancy.

b. 5 Rem whole body for the first 2 hours of the accident.

c. 25 Rem whole body for a 200 man-day continuous occupancy.

d. 25 Rem whole body for the first 2 hours of the accident.

ANSWER: 84 21428

a. 5 Rem whole body for a 200 man-day continuous occupancy.

Explanation:

The CREF System is designed to maintain the control room environment for a 200 man day continuous occupancy after a DBA without exceeding 5 rem whole body dose or its equivalent to any part of the body.

Distractors:

b. is incorrect because the 2 hour reference is incorrect. The candidate that confuses the 2 hour language from 10CFR100 may choose this answer.
c. is incorrect because 25 rem is greater than the design value.
d. is incorrect because 25 rem is greater than the design value and the design time is 200 man-days

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
85	21429	00	06/25/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320130, Recognize indications that require entry into emergency/abnormal procedures.

Related Lessons	
INT0320107	ABNORMAL CONDITION PROCEDURES
INT0320130	CNS Abnormal Procedures (RO) High Radiation

Related Objectives	
INT03201070000100	Given a list of symptoms, identify the abnormal condition

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6) (4.0/4.3)
LINK ONLY TO EOP/AOP LESSONS/QUESTIONS	

QUESTION: 85 21429 (1 point(s))

The plant was operating at power when an accident occurred that resulted in the following conditions:

- ➢ Radiation release rate has increased to twice the ODAM limit.
- ➢ Annunciator R-2/A-5, REACTOR BLDG PUMP ROOM HIGH TEMP is alarming.
- ➢ Reactor Building Exhaust Plenum radiation level is 7 mrem/hr.
- ➢ Refuel floor radiation levels are 5 mrem/hr.
- ➢ Refuel Floor CAM is alarming.
- ➢ Reactor Building D/P is 0"H2O.

What procedure(s) is/are required to be entered?

EOP-5A…

a. only.

b. and 5.1 RAD only.

c. and 2.4HVAC only.

d. 2.4HVAC and 5.1RAD.

ANSWER: 85 21429

d. 2.4HVAC and 5.1RAD.

EOP 5A entry is required by the reactor building low D/P. 5.1RAD is required to be entered due to the CAM alarm and 2.4HVAC entry is required by high area temperature alarm.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
86	21262	00	10/16/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	5	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070506, HPCI and one Loop of LPCI inop

Related Lessons	
INT0070506	OPS Tech. Spec. 3.5, Emergency Core Cooling (ECCS) and Reactor Core Isolation Cooling (RCIC) System

Related Objectives	
INT00705060010100	Given a set of plant conditions, recognize non-compliance with a Section 3.5 LCO.
INT00705060010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.5 LCO, determine the ACTIONS that are required.

Related References	
3.5.1	ECCS Operating

Related Skills (K/A)	
2.2.22	Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2) (3.4/4.1) **EXAM USE ONLY**

QUESTION: 86 21262 (1 point(s))

The plant is at 100% power. The following conditions then occur:

- 1200 on 10/16, HPCI is declared inoperable
- 1200 on 10/17, LPCI loop A is declared inoperable.

When is the plant **first required** to be in mode 3 by Technical Specifications?

a. 0100 on 10/18.

b. 0000 on 10/21.

c. 0000 on 10/24.

d. 1200 on 10/30.

ANSWER: 86 21262

b. 0000 on 10/21.

Since HPCI and one LP ECCS system are inoperable condition D of TS3.5.1 is entered. This starts a 72 hour clock to restore one of the systems to operable. At the end of the 72 hours Condition G is entered which now directs 12 hours to be in mode 3. 72 hours plus 12 hours from 1200 on 10/17 is 0000 on 10/21.

a. is incorrect because at this time the plant is not yet required to be in mode 3. The examinee that mistakenly enters 3.0.3 would choose this answer.

c. is incorrect because the plant is required to be in mode 3 prior to this time. The examinee that mistakenly enters only 3.5.1.A and then 3.5.1.B would choose this answer.

d. is incorrect because the plant is required to be in mode 3 prior to this time. The examinee that only entered Action C would choose this answer.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
87	21172	01	09/11/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070502, Predict the immediate effect of a loss of heaters on SLC and determine required actions.

Related Lessons	
INT0070502	CNS Tech. Spec. 3.1, Reactivity Control Systems

Related Objectives	
INT00705020010100	Given a set of plant conditions, recognize non-compliance with a Section 3.1 LCO.

Related References	
2.2.74	Standby Liquid Control System
6.LOG.601	Daily Surveillance Log - Modes 1, 2, and 3
3.1.7	Standby liquid control (SLC) system
3.1.7-1	Technical Specification LCO Figure
3.1.7-2	Technical Specification LCO
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

Related Skills (K/A)	
211000.A2.05	Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences...: (CFR: 41.5/45..5) Loss of SBLC tank heaters (3.1/3.4)

QUESTION: 87 21172 (1 point(s))

With the plant operating at 100% power, the following occur:

08:00 9-5-2/F-7 SLC TANK HEATER GROUND TO SOLUTION alarms.
08:20 The heater is deenergized to support troubleshooting.
09:30 9-5-2/G-8 SLC TANK HI/LOW TEMP alarms.
09:31 SO reports SLC Boron Solution Temp at 85°F and Tank Volume at 78%.
09:31 Chemistry sample indicates SLC Boron Solution Concentration at 15.4 weight percent pentaborate.

What impact does this have on SLC operability?
By procedure, what action is appropriate next?

a. Both SLC subsystems are inoperable.
Enter 8 hour LCO.

b. Both SLC subsystems are inoperable.
Enter a 7 day LCO.

c. Both SLC subsystems are operable.
Install heaters in the area to maintain SLC operability.

d. Both SLC subsystems are operable.
Partially drain and refilled the SLC tank with demin water to maintain SLC operable.

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ANSWER: 87 21172

c. Both SLC subsystems are operable.
Install heaters in the area to maintain SLC operability.

Provide the candidate with Figures 3.1.7-1

Per 2.2.74 11.1.2, have maintenance install portable heating in the SLC System area to raise ambient temperatures.
Distracter a, b. Both SLC subsystems remain operable. 78% tank level equates to 3560.8 gallons. Per TS Figure 3.1.7-1, concentration at 15.4% with volume at 3560.8 gallons is in the ACCEPTABLE region. Per TS Figure 3.1.7-2, temp at 85°F with concentration at 15.4% is in the ACCEPTABLE region. If the examinee incorrectly calculates tank volume in gallons, they could determine unacceptable SLC parameters are present when they are not.
Distracter d. Although an acceptable action in 2.2.74, section 11. This action would place the SLC solution closer to unacceptable parameters by diluting the concentration and lowering the SLC solution temp with colder demin water.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
88	21430	00	07/09/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070504, Knowledge of TS Bases

Related Lessons
INT0070504 CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010200	Discuss the applicable Safety Analysis in the Bases associated with each Section 3.3 Specification.

Related References	
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

Related Skills (K/A)	
2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CFR: 43.2) (2.5/3.7)
LINK ONLY TO TECH SPEC LESSONS & QUESTIONS	

QUESTION: 88 21430 (1 point(s))

What is the Technical Specification basis for having the Intermediate Range Monitor Neutron Flux-High trip?

a. Provides protection against the control rod drop accident by limiting fuel enthalpy to less than 280 cal/gm fuel damage limit.

b. Provides protection against local control rod withdrawal errors to limit peak fuel enthalpy below 170 cal/gm fuel failure threshold criterion.

c. Provides protection against a single sequence error that results a high local flux to ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

d. Provides protection against violation of the MCPR Safety Limit and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event.

ANSWER: 88 21430

b. Provides protection against local control rod withdrawal errors to limit peak fuel enthalpy below 170 cal/gm fuel failure threshold criterion.

The IRM provides mitigation of the neutron flux excursion. To demonstrate the capability of the IRM System to mitigate control rod withdrawal events, generic analyses have been performed (Ref. 3) to evaluate the consequences of control rod withdrawal events during startup that are mitigated only by the IRM. The continuous rod withdrawal during reactor startup analysis (Refs. 2 and 3), which assumes that one IRM channel in each trip system is bypassed, demonstrates that the IRMs provide protection against local control rod withdrawal errors and results in peak fuel enthalpy below the 170 cal/gm fuel failure threshold criterion.

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specifications knowledge of Technical Specification bases are required to make these assessments.

Source: Direct PTM21309

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
89	21431	00	07/09/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Administrative	INT0320211, Predict the impact of a stuck detector and determine required actions

Related Lessons	
INT0320211	INSTRUMENTATION OPERATING PROCEDURES (SRO)

Related Objectives	
INT03202110000100	Given the procedure, discuss any Notes, Limitations, Precautions, and interlocks that pertain to operator actions
INT03202110000200	Given the appropriate procedure discuss how and when control rod coupling verification is to be performed
INT03202110000300	Identify the signature and logging requirements for temporary setpoint changes. (4.0.1)
INT03202110000400	Describe when a temporary setpoint change may be required and who must be informed. (4.0.1)

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
215004.A2.03	Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those...: (CFR: 41.5 / 45.6) Stuck detector (3.0/3.3)

QUESTION: 89 21431 (1 point(s))

A plant startup is in progress with the following conditions:

- IRM C and IRM F are failed downscale and are bypassed.
- SRM B is downscale and has **not** been bypassed.
- The shorting link switches are in OPEN.
- Power level is on IRM range 3 for all operable IRMs.
- The SRMs counts are approaching 10^5 cps.

As the crew attempts to withdraw the SRMs, SRM A detector fails to move. The power ascension is stopped and maintenance determines that the detector is stuck and cannot be freed.

1.) If the power ascension is continued in this condition, what effect would the stuck detector have on operation?

2). What actions are required?

a. Rod block and Full Scram.
Enter procedure 4.1.1, bypass SRM channel A **and** deenergize SRM A by removing drawer fuses after the mode switch is in RUN.

b. Rod block only.
Immediately discontinue rod withdrawal, enter procedure 2.1.4 and shutdown the reactor.

c. Rod block and Full Scram.
Immediately discontinue rod withdrawal, enter procedure 2.1.4 and shutdown the reactor.

d. Rod block only.
Enter procedure 4.1.1, bypass SRM channel A **and** deenergize SRM A by removing drawer fuses after the mode switch is in RUN.

ANSWER: 89 21431

a. 1). Rod block and Full Scram.
2). Enter procedure 4.1.1, bypass SRM channel A **and** deenergize SRM A by removing drawer fuses after the mode switch is in RUN.

Explanation:

If power is raised with SRM A detector stuck and unbypassed with the shorting link switches open a rod block and full scram would occur. 4.1.1 provides directions to bypass the stuck detector. Since IRMs are on range 4 Technical specifications allow the continued power ascent. When the mode switch is in RUN, 4.1.1 requires that the SRM A fuses in the SRM drawer be pulled.
:

Distractors:

b. is incorrect because a full scram would also occur and continued rod withdrawal is allowed.
c. is incorrect because continued rod withdrawal is allowed.
d. is incorrect because a full scram would occur if power is raised before any other actions are taken.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
90	21432	00	07/10/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	2	1	3	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320131, Predict the impact of over current on breakers and determine the procedures to enter to mitigate the consequences.

Related Lessons	
INT0320131	CNS Abnormal Procedures (RO) Electrical

Related Objectives	
INT0320131T0T0100	Given plant condition(s), determine from memory any automatic actions listed in the applicable Abnormal/Emergency Procedure(s) which will occur due to the event(s).
INT0320131S0S0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
262001.A2.10	Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences...: (CFR: 41.5 / 45.6) Exceeding current limitations (2.9/3.4)

QUESTION: 90 21432 (1 point(s))

Given the following conditions:

- ➢ Reactor is in Hot Shutdown.
- ➢ The emergency transformer is deenergized.
- ➢ DG 1 is paralleled to 1F.
- ➢ The diesel governor fails to maximum fuel rack position (DG current rises to greater than 180% of normal full load current.)

1) What breaker 1FA or 1AF trips in response (either direct or indirect) to the high DG1 current?
2) What action would be required following the trip?

a. **ONLY** 1AF trips.
Enter 5.3EMPWR

b. **ONLY** 1FA trips.
Enter 5.3EMPWR

c. **ONLY** 1AF trips.
Enter 5.3AC480

d. **ONLY** 1FA trips.
Enter 5.3AC480

ANSWER: 90 21432

d. **ONLY** 1FA trips.
Enter 5.3AC480

Explanation:

1FA is tripped by the over current condition. 1AF will not trip because 1AF is in NORMAL AFTER CLOSE and Bus 1A is energized. Once 1AF trips the DG will overspeed and trip Deenergizing 4160 1F. Since 4801F is lost entry into 5.3AC480 is required. The entry conditions for 5.3EMPWR are not met because 4160A, B an E remain energized.

REFERENCES: STCOR001-01-02
PR 2.2.18, page 13, 15, section 4.6.2, 4.9.2.4
PR 2.2.20, page 16, section 4.9

Distractors

a. is incorrect because 1FA trips on over current and 5.3EMPWR is not entered.
b. is incorrect because 5.3EMPWR is not entered.
c. is incorrect because 1FA trips on over current.

Source: Modified 19223

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
91	21433	00	07/09/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070513, Determine conditions that are entry level TS

Related Lessons
INT0070513 CNS Tech. Spec. 5.0, Administrative Controls

Related Objectives	
INT00705130010100	Given a set of plant conditions, recognize non-compliance with a Chapter 5.0 Requirement.
INT00705130010200	Given a set of conditions that constitutes non-compliance with a Chapter 5.0 Requirement, determine the actions that are required.

Related References	
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

Related Skills (K/A)	
2.1.33	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3) (3.4/4.0) **EXAM USE ONLY**

QUESTION: 91 21433 (1 point(s))

Two temporary bladder tanks have been installed as a temporary surge volume for radwaste. The bladders are located inside the protected area but have no retaining walls. Radwaste effluent is currently being discharged to bladder tanks A and B.

Chemistry report that following data following a sample of the tanks:

- Tank A contains 5,000 gallons and 12 Ci of activity (excluding tritium and noble gas).
- Tank B contains 6,000 gallons and 9 Ci of activity (excluding tritium and noble gas).

What action(s) is/are required?

a. Immediately suspend addition of radwaste to bladder tank A.

b. Immediately suspend addition of radwaste to both bladder tanks.

c. Immediately initiate action to reduce the curie content of both tanks.

d. Shutdown per TS LCO 3.0.3.

ANSWER: 91 21433

a. Immediately suspend addition of radwaste to bladder tank A.

<u>Provide the Candidate with TS5.5.8 and D3.1.4.</u>

Explanation:

Tanks without retaining walls are limited to 10 Ci each. In this case the tank A is greater than 10 curies. IAW DLCO 3.1.4 Action A Immediately suspend the addition of radioactive material to the tank.

Distractors:

b. is incorrect because tank B is lower than the 10 Ci limit.

c. is incorrect because tank B has no requirement to reduce the Ci content of the tank.

d. is incorrect because no 3.0.3 shutdown is not required. DLCO3.0.3 is the applicable vehicle for non-compliance with ODAM limits.

Source New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
92	16634	00	10/20/1997	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Administrative	COR0010502, FIRE PROTECTION TS Required actions

Related Lessons
COR0010502 FIRE PROTECTION SYSTEM

Related Objectives	
COR00105020010200	Given condition(s) and/or parameters associated with the Fire Protection system, determine if related Technical Requirements Manual Limiting Condition for Operation are met.

Related References	
0.23	CNS Fire Protection Plan
6.FP.101	Fire Pump 31 Day Operability Test
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

Related Skills (K/A)	
2.2.22	Knowledge of limiting conditions for operations and safety limits. (CFR: 43.2 / 45.2) (3.4/4.1) **EXAM USE ONLY**

QUESTION: 92 16634 (1 point(s))

Following the performance of 6.FP.101, Fire Pump 31 Day Operability Test, it is noted that Electric Fire Pump C would NOT automatically start and would NOT start when the START pushbutton was pressed on local control panel.

What is action is the required action?

a. Restore the fire pump to OPERABLE status within 7 days.

b. Initiate a Fire Protection System impairment for the fire pump.

c. Establish an hourly Fire Watch patrol in the affected areas within 1 hour.

d. Establish another fire suppression water system as a backup within 24 hours.

ANSWER: 92 16634

b. Initiate a Fire Protection System impairment for the fire pump.

"C" fire pump is NON-TRM. Only action is to initiate a fire impairment. There are no specific actions specified in 0.23 for the impairment.

REFERENCE: TRM: T 3.11.2, TLCO
6.FP.101: Step 7.1
0.23: 6.2, 6.4, and Attachment 1

Distracter a:
T3.11.2 TLCO statement is still met. Conditions and actions are not entered.
Distracter d:
Actions if certain detection systems become inoperable.
Distracter a:
Correct response if E fire pump or diesel fire pump inoperable and not restored within 7 days or if the system is inoperable for other reasons than an inoperable fire pump (E or diesel).

Provide to Candidate: TLCO 3.11.2

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
93	21435	00	07/09/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Administrative	INT0070601, Determine consequences of thermal limit out of limits and determine required actions when thermal limit is exceeded.

Related Lessons	
INT0070601	TRM - Overview, Reactor Power Distribution, Reactor Coolant and Refueling

Related Objectives	
INT0070601001040A	Given plant conditions and the TRM, determine ACTIONS required per the following TLCOs: T.3.2.1 Linear Heat Generation Rate
INT0070601001020A	Discuss the applicable Bases associated with each of the following TRM Limiting Conditions for Operation (TLCOs): T.3.2.1 Linear Heat Generation Rate
INT0070601001030A	Given plant conditions, determine if the following TLCOs are met: T.3.2.1 Linear Heat Generation Rate

Related References	
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

Related Skills (K/A)	
290002.A2.05	Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of...: (CFR: 41.5 / 45.6) ?Exceeding thermal limits (3.7/4.2)

QUESTION: 93 21435 (1 point(s))

The plant is operating at 95% power when a loss of feedwater heating occurs. Feedwater temperature drops several degrees and reactor power increases to 98%. A Gardel Periodic case is demanded and the following were noted for the most limiting core locations:

MAPRAT	0.992
MFLPD	1.001
MFLCPR	0.988

What potential consequence could result from continued operation with these conditions?

What action is required?

a. Fuel clad cracking due to high stress.
Immediately initiate action to restore operation to within limits.

b. Fuel clad cracking due to lack of cooling.
Restore operation to within limits within 2 hours.

c. Fuel clad cracking due to high stress.
Restore operation to within limits within 2 hours.

d. Fuel clad cracking due to lack of cooling.
Immediately initiate action to restore operation to within limits.

ANSWER: 93 21435

a. Fuel clad cracking due to high stress.
Immediately initiate action to restore operation to within limits.

<u>Provide TLCO 3.2.1, TS 3.2.1 and TS 3.2.2.</u>

EXPLANATION OF ANSWER: With MFLPD > 1.0, LHGR is exceeding its limit. TLCO 3.2.1 requires immediate action to restore operation to within limits.

Distractors:

b. is incorrect because the failure mode when LHGR is exceeded is high stress. and immediate action is required by TLCO 3.2.1. .

c. is incorrect because with LHGR out of limits TLCO 3.2.1 requires entry..

d. is incorrect because the failure mechanism for LHGR exceeding its limit is clad cracking due to high stress.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
94	5574	04	07/02/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070601, TRM Chemistry - 7% power ascension continue?

Related Lessons	
INT0070601	TRM - Overview, Reactor Power Distribution, Reactor Coolant and Refueling

Related Objectives	
INT00706010010100	Explain the application of Improved Technical Specifications ITS sections 1.0 and 3.0 to the CNS Technical Requirements Manual.
INT0070601001020B	Discuss the applicable Bases associated with each of the following TRM Limiting Conditions for Operation (TLCOs): T 3.4.1 RCS Chemistry
INT0070601001030B	Given plant conditions, determine if the following TLCOs are met: RCS Chemistry
INT0070601001040B	Given plant conditions and the TRM, determine ACTIONS required per the following TLCOs: T.3.4.1 RCS Chemistry

Related References	
2.1.1	Startup Procedure
T3.4.1	RCS Chemistry
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.

Related Skills (K/A)	
2.1.34	Ability to maintain primary and secondary plant chemistry within allowable limits. (CFR: 41.10 / 43.5 / 45.12) (2.3/2.9)

QUESTION: 94 5574 (1 point(s))

The plant is in the process of a reactor startup, at approximately 7% power when the following occur:

- ➢ At 08:00 on 7/2 the Reactor Coolant Continuous Conductivity Monitor indicates that conductivity is 2.3 micro mho/cm and lowering slowly.

Does Procedure 2.1.1 allow the power ascension to continue?

How long is allowed to restore conductivity to within limits?

a. No. 48 hours

b. No. 72 hours

c. Yes. 48 hours

d. Yes. 72 hours

ANSWER: 94 5574

a. No. 48 hours

Provide TRM Table 3.4.1-1 to the Candidate.

TRM 3.4.1 Chemistry limits are exceeded for condition 2 of Table 3.4.1-1, $\leq$ 10% power and > 212°F and CONDITION A requires restoring RCS Chemistry to within limits with a completion time of 48 hours from the time the limit was exceeded. The note in the actions section states TLCO 3.0.4 is not applicable so the power ascension could continue, however, Procedure 2.1.1 directs you to hold the startup until chemistry is within spec.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
95	19366	00	09/15/2003	10/07/2006	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	Multiple Choice	

Topic Area	Description
Administrative	INT0231001, S/D Risk Mngt - determine which activity can continue

Related Lessons	
INT0231001	OPS Shutdown Risk Management

Related Objectives	
INT02310010000300	Given specific plant configurations, determine if any deviations from outage guidelines exist

Related References	
0.50	Outage Management Program

Related Skills (K/A)	
2.2.18	Knowledge of the process for managing maintenance activities during shutdown operations. (CFR: 43.5 / 45.13) (2.3/3.6)

QUESTION: 95 19366 (1 point(s))

The plant is in day 21 of a scheduled outage with the following conditions.

- RHR loop A is operating in shutdown cooling with RHR pump A.
- RHR pump C is inoperable due to a breaker malfunction discovered last shift.
- The plant is in MODE 5 with the fuel pool gates removed.

The following activities are scheduled to continue through the upcoming shift:

- Cleaning and inspection of MCC-R.
- 4160 BKR EG2 removal for inspection and repair.

Which of the following activities scheduled to be worked must be postponed?

a. 250 VDC B bus maintenance.

b. Replacement of the SLC squib valves.

c. Replacement of the CS Pump B breaker.

d. Replacement of CRD drive filter housings.

ANSWER: 95 19366

a. 250 VDC B bus maintenance.

250 VDC B should not be worked at this time. RHR-MO-18 is de-energized due to the MCC-R work and 250 VDC B powers RHR-MO-17. PCIS requirements would not be met.

b. No backup reactivity management method is required at this time.
c. Replacement of the CS-B breaker would be allowed at this time.
d. CRD can be worked at this time because adequate inventory control systems are available.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
96	20517	00	08/06/2005	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	3	Multiple Choice	

Topic Area	Description
Administrative	Knowledge of the refueling process.

Related Lessons
NONE

Related Objectives
NONE

Related References	
10.25	Refueling - Core Unload, Reload, and Shuffle
10CFR55.43 (7)	Fuel handling facilities and procedures.

Related Skills (K/A)	
2.2.32	Knowledge of the effects of alterations on core configuration. (CFR: 43.6) (2.2/3.3)

QUESTION: 96   20517 (1 point(s))

The reactor has been shutdown for 7 days with core fuel shuffling activities in progress. During the shuffle two (2) bundles are maintained around each operable SRM. In addition to the bundles around the SRMs, additional fuel is maintained in the core during the shuffle.

What purpose does this additional fuel serve during the shuffle?

a.   Ensure the cooling capacity of the FPC system is not exceeded.

b.   Ensures maintenance of the shutdown K_{eff} of the Spent Fuel Storage Pool (SFSP).

c.   Provide neutronic coupling of the core to ensure indication of neutron population at the SRM detectors.

d.   Limits the I-131 inventory in the Spent Fuel Storage Pool (SFSP) so that the limits of 10CFR100 would not be exceeded during a long duration loss of Fuel Pool Cooling.

ANSWER: 96 20517

c.   Provide neutronic coupling of the core to ensure indication of neutron population at the SRM detectors.

Reloading sequences are designed to ensure proper neutron flux monitoring. During core shuffle activities, at least two fuel bundles shall remain around each operable SRM with sufficient fuel remaining in the core at proper locations to ensure adequate core coupling.

Distractors:

a.   is incorrect because sufficient cooling capacity exists with RHR to cool the pool if the core were discharged.

b.   is incorrect as the spent fuel pool would have remain shutdown with the entire core offloaded.

d.   is incorrect as this is not a consideration for leaving fuel in the vessel.

Repeat from the 2005 NRC Exam

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
97	20551	0	04/29/2004	10/07/2006	Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	3	Multiple Choice	

Topic Area	Description
Emergency Plan	Authorization of Dose in Excess of 10CFR20 Limits

Related Lessons
GEN0030401 Emergency Plan for Licensed Operators

Related Objectives	
GEN0030401E0E0100	State the major differences in 10CFR20 and EPA-400 derived TEDE values.
GEN0030401F0F1200	Discuss precautions and limitations of 5.7.17, EMERGENCY RADIATION EXPOSURE CONTROL.
GEN0030401F0F1300	From memory, given conditions, determine if Emergency Exposures should be authorized, and if so, for whom.

Related References	
5.7.12	Emergency Radiation Exposure Control

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 97 20551 (1 point(s))

What minimum level of authority can authorize radiological exposures in excess of 10CFR20 limits? (Select the **first position** that has the authority to authorize the exposure).

a. Control Room Supervisor (non-emergency)

b. Shift Manager (non-emergency)

c. Emergency Director (emergency)

d. Radiological Manager (emergency)

ANSWER: 97 20551
c. Emergency Director

Only the Emergency Director has the authority to authorize exposures in excess of occupational limits.

Distractors:

a. is incorrect because the only the emergency director can authorize exposures in excess of occupational limits. Although in a very rare circumstance the Control Room Supervisor may act as the emergency director, the question asked what position could authorize the exposure and the only position that can authorize the exposure is the Emergency Director.

b. is incorrect because the only the emergency director can authorize exposures in excess of occupational limits. Although initially the Shift Manager will be the emergency director, the position of shift manager cannot authorize the exposure.

d. is incorrect because only the emergency director can authorize the exposure.

SRO Justification: SRO personnel fulfill the role of emergency director initially during an emergency, ROs cannot perform this function.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
98	20526	0	04/02/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Administrative	Liquid Release Authorization/Approval and Actions for a Lost CW Pump During Release

Related Lessons	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115B0B0100	State who, by title, authorizes releases of radioactive liquid effluents from CNS.
INT0320115B0B0300	State the number of Circulating Water Pumps required to be in service during liquid radioactive discharges.

Related References	
8.8.11	Liquid Radioactive Waste Discharge Authorization
10CFR55.43 (4)	Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Related Skills (K/A)	
2.3.3	Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g. / waste disposal and handling systems). (CFR: 43.4 / 45.10) (1.8/2.9)

QUESTION: 98   20526 (1 point(s))

The plant is operating at low power with 2 Circulating Water pumps running. **De-icing** is in progress. The Radwaste Operator indicates that the Floor Drain Sample Tank requires discharging.

1) Whose approvals/authorizations is/are required in order to accomplish this discharge?
2)   If one of the two operating circulating water pumps trip during the discharge, what action, if any, is required and why?

a. Chemistry department authorizes the release and the duty Shift Manager approves the release.
   Continue the discharge sufficient dilution flow exists.

b. Duty Shift Manager authorizes and approves the release.
   Continue the discharge sufficient dilution flow exists.

c. Chemistry department authorizes the release and the duty Shift Manager approves the release.
   Terminate the discharge insufficient dilution flow exists.

d. Duty Shift Manager authorizes and approves the release.
   Terminate the discharge insufficient dilution flow exists.

ANSWER: 98 20526

c. Chemistry department authorizes the release and the duty Shift Manager approves the release.
   Terminate the discharge insufficient dilution flow exists.

Answer c is correct because procedure 8.8.11 requires that chemistry authorizes the release and the duty Shift Manager approves the release. The loss of one CW pump would reduce flow to less than the minimum required and the discharge should be terminated.

Distractors:
a. is incorrect because the discharge should be terminated.
b. is incorrect because the discharge should be terminated and the chemistry department authorizes the release.
d. is incorrect because chemistry authorizes the release.

SRO Justification: This is an SRO only item because in accordance with procedure 8.8.11 only the duty Shift Manager can approve liquid radioactive releases.

Source: Direct

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
99	21436	00	06/25/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	5	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320107, Knowledge of abnormal condition procedures.

Related Lessons	
INT0320107	ABNORMAL CONDITION PROCEDURES

Related Objectives	
INT03201070000100	Given a list of symptoms, identify the abnormal condition
INT03201070000200	Given a specific procedure title, describe the automatic actions listed in the procedure
INT03201070000300	Given a specific procedure title, or adequate information of plant conditions and indications, analyze the immediate actions required
INT03201070000400	Given a specific procedure title, appraise the key concepts from the discussion section
INT03201070000500	Given a specific procedure, explain or analyze any NOTES and CAUTIONS addressed in the procedure

Related References	
10CFR55.43 (2)	Facility operating limitations in the technical specifications and their bases.
10CFR55.43 (5)	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Related Skills (K/A)	
2.4.11	Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13) (3.4/3.6)

QUESTION: 99 21436 (1 point(s))

A plant startup is in progress with power at 19%. The following data is noted:

- ➢ Reactor water conductivity is 0.5 µmho (RWCU-CR-132, REACTOR WATER CONDUCTIVITY)
- ➢ Reactor water sulfates are 225 ppb.
- ➢ Reactor water chloride is 25 ppb.
- ➢ Condensate F/D combined outlet reaches is 2.2 µmho on CF-CR-25 (Channel 2)

What procedure(s) are required to be entered?
Why is action required for these plant conditions?

a. 2.4CHEM and 2.1.5 Reactor Scram
Combined F/D outlet conductivity requires immediate action to prevent degradation of Reactor Water Chemistry.

b. 2.4CHEM ONLY
High sulfates increase probability of Inter-granular Stress Corrosion Cracking

c. 2.4CHEM ONLY
Combined F/D outlet conductivity requires immediate action to prevent degradation of Reactor Water Chemistry.

d. 2.4CHEM and 2.1.5 Reactor Scram
High sulfates increase probability of Inter-granular Stress Corrosion Cracking

ANSWER: 99 21436

ANSWER:

a. 2.4CHEM and 2.1.5 Reactor Scram
Combined F/D outlet conductivity requires immediate action to prevent degradation of Reactor Water Chemistry.

Answer source: 2.4CHEM

The High sulfates require that entry into 2.4CHEM. IAW 2.4CHEM if Combined Filter Demineralizer conductivity reaches 2.0 µmho enter 2.1.5. This prompt action prevents a more rapid degradation of reactor water chemistry.

Distractors:

b. is incorrect because entry into 2.1.5 is also required because combined F/D conductivity is high.
c. is incorrect because entry into 2.1.5 is also required because combined F/D conductivity is high.
d. is incorrect because the high sulfates do not require entry into 2.1.5.

Source: New

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
100	19935	02	03/17/2004	10/07/2006	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	2	1	8	Multiple Choice	

Topic Area	Description
Technical Requirements Manual	INT0070509, Cascade from 3.8.1 to PAM

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications
INT0070509	OPS Tech. Spec. 3.8, Electrical Power Systems
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.
INT00705090010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.8 LCO, determine the ACTIONS that are required.
INT00705010010200	Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.

Related References	
3.8.1	AC Sources - Operating
3.3.3.1	Post accident monitoring (PAM) instrumentation

Related Skills (K/A)	
2.4.3	Ability to identify post-accident instrumentation. (CFR: 41.6 / 45.4) (3.5/3.8)

QUESTION: 100 19935 (1 point(s))

The plant was operating at rated power when the following occurred:

- NBI-LI-85A (Wide Range RPV water level) becomes inoperable at 0900 on 7/06.
- An air leak in the starting air system for DG2 occurs at 1200 on 7/12. Air pressure in both receivers lowers to 100 psig.

IF conditions do not change, what is the **EARLIEST** date and time that Technical Specifications requires the plant to enter **MODE 3**?

a. 1600 on 7/12.

b. 1600 on 7/17.

c. 2400 on 7/19.

d. 0400 on 7/20.

ANSWER: 100 19935

c. 2400 on 7/19.

<u>Proved 3.8.3 and 3.3.3.1 to the candidate.</u>

3.8.3.F requires DG2 be declared inoperable immediately.

NBI-LI-85A (Wide Range RPV water level PAM instrument powered from CCP-1A) is inoperable. DG #2 becomes inoperable requiring 4 hours later, the Conditions and Required Actions for both Wide Range RPV water level PAM instruments inoperable (3.3.3.1 Condition "C") must be entered as the inoperable PAM instrument on a division opposite that of the inoperable DG. NBI-LI-85B is powered by CCP which is supported by DG #2. NBI-LI-85A is "an inoperable redundant required feature supported by the other DG".

Enter 3.3.3.1.C at 1600 on 7/12. Enter 3.3.3.1.D at 1600 on 7/19. Enter 3.3.3.1.E at 1600 on 7/19. Be in MODE 3 0400 on 7/20
DG is more limiting (3.8.1). Be in MODE 3 by 2400 on 7/19

Source: Direct